Westinghouse Technology Systems Manual

Chapter 16

RADIATION MONITORING SYSTEM

Section

Westinghouse Technology Systems Manual

Section 16.0

Radiation Monitoring System

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16.0 RADIATION MONITORING SYSTEM

Learning Objectives:

1. State the purposes of the radiation monitoring system.

2. List the two classes of radiation monitors and give four examples of each.

3. List four radiation monitors that provide automatic actions (other than alarms) and briefly describe the action provided.

4. List and briefly describe the two types of failed fuel monitors.

5. List the radiation monitors which identify the following:

   a. Primary to secondary leakage
   b. Primary to containment leakage

16.1 Introduction

The purposes of the radiation monitoring system (RMS) are as follows:

1. Continuously monitor radiation levels of various plant areas, processes and effluents.

2. Provide alarms and/or automatic actions if preset limits are exceeded.

The radiation monitoring system is divided into the following: process radiation monitoring (PMS), explained in Section 16.2.1, and area radiation monitoring (ARM), explained in Section 16.2.2.

16.2 System Description

16.2.1 Process Radiation Monitoring System

The PRM monitors the radiation level of various process liquid and gas streams that may serve as discharge routes for radioactive materials. These monitors are provided to indicate the radioactivity of the process stream and to alert operating personnel when operational limits are approached for the normal release of radioactive material to the environment.

On process streams that do not discharge to the environs, such as the component cooling water system (Section 14.1), process monitors are provided to indicate process stream malfunctions. This is accomplished by detecting the normal background radiation of the system and by alerting the operator with an annunciator if an accumulation of radioactive material occurs in the system. In addition to providing continuous indication and alarms, the PRM may provide various automatic functions, such as the closing of vent valves, discharge valves, etc.

If the activity level in the process stream reaches a predetermined setpoint, the system will perform its automatic function, which ensures that the discharge of radioactive material to the environs is limited. The PRM can monitor its process by one of two methods. These methods are called in-line monitors and off-line monitors.

IN-LINE monitor

An in-line monitor system has the detector probe directly immersed in the process stream. The advantage of this type of monitoring is that the detector probe will be provided a representative sample of the process and responds rapidly to activity changes. The disadvantages of this system are that if the process has a turbulent flow the detector probe must be protected by placing it in a well, which lowers the sensitivity of the probe, and if the probe fails, the system must either be secured or a means of bypassing flow around the probe must be provided (only for directly immersed probes).

OFF-LINE monitor

An off-line monitor system contains piping, valves, detector probes (usually two in parallel), and a motive force device, such as a pump or a fan. This system will take a suction on the process stream, pass the flow to the detector, and then return the sample to the process stream. The advantages of this system are that with lower flow rates, the detector probe can be directly immersed into the sample stream without a protection well, therefore increasing probe sensitivity, and if a detector fails, it can be isolated (by inlet and outlet valves), and the process stream is not affected.

The disadvantages of this system are that it may not be receiving a representative sample of the process stream and may not be as responsive to rapid changes of activity in the process. At the time of this writing there is no preferred method of process monitoring. Both types, IN-LINE and OFF-LINE monitors, are found throughout the industry.

16.2.2 Area Radiation Monitoring System

The ARM are located in selected areas of the plant, such as the Control Room, Containment Building, and various areas or rooms in the Auxiliary Building. The purposes of the area monitors are to give continuous indication of background radiation and to alert plant personnel of high radiation. The alarms associated with area radiation monitors include annunciators in the control room area and local alarms. The local alarms are at the radiation detectors and provide indication of high radiation both visually (flashing light) and audibly (horn or buzzer).

16.3 Component Description

16.3.1 Radiation Detectors

There are various types of detectors used in both the process and area radiation monitoring systems. Some of the most commonly used detectors will be explained in this section.

Geiger-Mueller Tube

The geiger-mueller (G-M) tube is a gas filled chamber with a center electrode as shown in Figure 16-1. The G-M tube operates by sensing an incident particle, such as a beta or gamma, interacting with the gas inside the chamber to produce a few ion pairs. As the ion pairs are accelerated towards the wall and electrode, they will produce more ions pairs until there are millions of ion pairs produced inside the chamber.

The affect of this mass production of ions is called avalanching and is the result of the high voltage potential between the chamber and the center electrode. When avalanching occurs, the millions of ions, both positive and negative, are collected on the chamber wall and center electrode. When this happens, a pulse is produced. The pulse is always the same size. Therefore, it is not possible to tell the type or energy of the incident particle from the pulse, but only that radiation is present.

This instrument will use a gas that is easily ionized, such as argon. However, when the positive ions are collected on the center electrode, secondary photons, in the form of ultraviolet light, are produced. These photons will then travel across the gas volume and interact with the walls of the chamber. This interaction produces electrons which starts the avalanche all over again. To prevent this undesired secondary avalanche, another gas is normally added to the chamber and mixed with the ionizing gas. This second gas is called a quenching gas and is normally a halogen, such as bromine or chlorine.

Once the avalanche occurs, the detector will be saturated with ions, and another incident particle entering the detector will not be seen. After the quenching takes place, the detector will be able to sense another incoming particle. Due to this time between avalanches and quenching, a G-M tube will only be useful in certain radiation fields. The problem with high radiation areas is that there are so many incident particles that the detector will stay saturated. When this occurs the output from the detector will go to zero, and the meter reading would be full downscale. To overcome this effect, the circuitry is designed so that the meter reading will read full scale if the detector becomes saturated.

Scintillation Detectors

The scintillation detector consists of a crystal, window, and a photo multiplier tube as shown in Figure 16-2.

The scintillation detector works on the principle that when a radioactive particle interacts with certain materials (crystal), light is produced. By measuring the light that is emitted, the energy and amount of the original radioactive particle can be determined. Scintillation detectors can accurately measure different types of radiation, such as alphas, betas, or gammas. This is accomplished by using different types of materials (crystals) for each type of radiation.

A scintillation detector is constructed such that it is sensitive to only one type of radiation, i.e., if beta and gamma radiation levels were to be measured in a process, both a beta scintillation and a gamma scintillation detector must be provided. The scintillation detector operates by a radioactive particle interacting with the crystal, which causes ionization of some of the atoms. These ionized atoms now have electrons in the excited state, but the excitation is not great enough for the electron to escape. Since these atoms are now in an excited state, they will radiate this excess energy in the form of photons (light rays).

The photons are transmitted from the sensing crystal to the photo multiplier tube via a quartz window. Quartz is used to transmit the light produced in the crystal because it will not distort the photons emitted from the crystal.

The photo multiplier tube consists of a photo cathode, dynodes, anode and outer chamber wall. The light produced by the crystal interacts with the photo cathode which then produces electrons. These electrons are then attracted to the first positively charged dynode. When an electron strikes the dynode several electrons will be produced (typically for each electron striking a dynode, two to four electrons will be produced).

The electrons produced by the dynode will then be directed to the next dynode for further multiplication. The result of this multiplication process is that generally about one million electrons are produced from each electron produced by the photo cathode. This large electron flow is then collected by the anode, and an electrical current is produced and measured by the circuitry. This circuitry can be setup so that the detectors output can be in counts per minute or as a dose rate in mrem or Rem per hour.

The scintillation detector is useful not only in detecting radiation, but also in laboratory work, since this detector can also measure the energy of the incident radiation. The energy can be measured because the pulse of electrons at the anode is proportional to the energy of the original incident radiation, i.e. the more energy the radioactive particle has, the more light that will be emitted from the crystal. This will in turn cause more electrons to be produced by the photo cathode, which after the electron multiplication, causes a larger pulse at the anode.

16.3.2 Plant Radiation Monitors

Containment Air Particulate Detector

The containment air particulate detector (Figure 16-4) is sufficiently sensitive for detection of reactor coolant leakage into the containment. This instrument is capable of detecting leakage rates of 5 cc/min. within minutes after the leak occurs.

Continuous air samples are taken from the containment atmosphere near the reactor containment fan cooler inlet, drawn outside the containment in a closed, sealed system, and monitored by a scintillation counter and movable filter paper detector assembly. The air sample is passed through a filter paper which collects 99% of all particulate matter greater than 0.3 microns in size. The constantly moving filter paper is viewed by the scintillation detector, which then transmits the activity level to the main control room.

The activity level is then indicated on a meter and on a recorder. The air sample, after passing through the detector, is returned to the containment. The detector is used principally to detect the following radioactive isotopes in the containment atmosphere: I-131, I-133, CS-134 and CS-137. Receipt of a high activity level in the containment will be annunciated in the control room. In addition to the annunciator, the following automatic actions will occur:

- The containment purge supply and exhaust dampers will close.
- The pressure and vacuum relief valves will close (if they are open).
- The containment fan cooler dampers shift to the accident mode (the fans, however, remain in fast speed).

There is a brief description of this system and the following Process Radiation Monitors in Table 16-1. The table lists the Process Monitor location, type of detector used, and the automatic actions provided by the monitor.

Containment Noble Gas Monitor

The containment gas monitor (Figure 16-4) is provided in order to supply the operator with information pertaining to the noble gas activity in the containment. This activity is due to neutron activation of the primary shield cooling air and from leaks in the reactor coolant system when operating with cladding defects in the fuel. Continuous samples are taken from the containment atmosphere. After the sample passes through the previously described air particulate monitor, it flows through a closed sealed system to the gas monitor assembly. The samples flow continuously to a fixed, shielded volume, where the activity is measured by a Geiger-Mueller tube or beta scintillation detector. The air sample is then returned to the containment. The activity level is sent to the main control room, where the level is indicated on a meter and on a recorder. The detector is used principally to detect the following noble gases: Kr-85, Ar-41, Xe-133 and Xe-135. Receipt of high activity will annunciate in the main control room. In addition to the annunciator, the following automatic actions will occur:

- The containment purge and exhaust dampers will close.
- The pressure and vacuum relief valves will close (if they are open).
- The containment fan cooler dampers shift to the accident mode (the fans, however, remain in fast speed).

The radio gas detector will supplement the information obtained from the air particulate monitor regarding the occurrence of leakage from the primary system.

Containment Purge Exhaust Monitor

This channel (Figure 16-4) monitors the effluent from the containment purge for gaseous activity, iodine, and particulate activity whenever the purge system is in operation. The system consists of three separate channels, one of which is a fixed filter air particulate monitor with a beta scintillation detector, the second is a Geiger-Mueller detector for monitoring gaseous activity, and the third is a spectrometer grade gamma scintillation detector for iodine monitoring.

Detector outputs are transmitted to the main control room where they provide indication on meters and recorders. High radioactivity during the containment purge operations will be annunciated in the control room. In addition to the alarm, the purge supply and exhaust dampers will automatically close.

Auxiliary Building Ventilation System Monitor

This channel (Figure 16-4) continuously monitors the ventilation system exhaust air from all the potentially contaminated equipment cubicles in the auxiliary building. This system uses one moving filter monitor for particulates and a fixed filter unit for iodine. The detector for particulates uses a moving filter paper and monitors this filter with a beta scintillation detector. For iodine monitoring, a spectrometer grade gamma scintillation detector is used.

The outputs from these detectors are transmitted to the main control room for indication on meters and recorders. High radioactivity from either detector will be annunciated in the main control room. Detection of high radioactivity from the iodine monitor will automatically realign the auxiliary building ventilation system so that the exhausts from the equipment cubicles will be routed through charcoal filter banks prior to exhausting to the atmosphere.

Plant Vent Stack Monitor

This channel (Figure 16-4) monitors the ventilation system air discharging from the auxiliary building ventilation system to the plant ventilation stack. The sample gas is returned to the suction of the auxiliary building exhaust fans. The channel utilizes four Geiger-Mueller tubes connected in parallel. The radioactivity is indicated by a meter and recorder located in the main control room. High radiation at this monitor will actuate an annunciator in the main control room. There are no automatic functions associated with this channel. This monitor is used principally to detect Kr-85, Ar-41, Xe-133, and Xe-135.

Control Room Intake Air Monitor

This channel continuously monitors the outside air intake to the control room. Two monitors, one for particulates and one for iodine, are provided. The particulate monitor uses moving filter paper and a beta scintillation detector, and the iodine monitor employs a fixed filter and gamma scintillation detector. The detector outputs are transmitted to the control room, where the radioactivity levels are indicated by meters and recorders. High radiation conditions are annunciated in the main control room. In addition to the annunciators, an alarm in either channel will cause the ventilation system air inlet to close, and makeup air, for maintaining a slightly pressurized control room, will be introduced from the turbine building.

Condenser Air Discharge Gas Monitor

This channel (Figure 16-4) receives a continuous air sample from the air ejector exhaust header, monitors it for gaseous radioactivity, and provides the plant operator with a rapid indication of a primary to secondary leak. The sample gas is returned to the gas effluent. This channel uses a G-M detector whose output is transmitted to the main control room for indication on meters and recorders. A high radiation condition will be annunciated in the main control room. There is no automatic function associated with this channel. This monitor is used principally to detect noble gases such as Kr-85, Xe-133, and Xe-135.

Steam Generator Blowdown Liquid Monitor

T his channel (Figure 16-4) monitors the liquid phase of the secondary side of the steam generator for radioactivity, which would indicate a primary to secondary system leak, providing backup information to that of the condenser air ejector gas monitor. Samples from each of the four steam generator bottoms are mixed in a common header and the common sample is continuously monitored by a scintillation counter and holdup tank assembly. Upon indication of high radioactivity, each steam generator is individually sampled in order to determine which unit is leaking. The detector output is transmitted to the main control room where indication is provided by both a meter and a recorder. A high radiation condition will be annunciated in the control room. There are no automatic functions associated with this channel. This monitor is used principally to detect Co-60.

Component Cooling Water System Monitor

This channel continuously monitors the component cooling water system (Section 14.1) for activity which would be indicative of a leak from one of the components that this system is cooling. A gamma scintillation detector is used for this monitor, and its output is transmitted to the control room. The activity level is indicated on a meter and a recorder. A high radiation condition will be annunciated in the control room. After receiving this alarm, the operator will isolate the affected component to stop the radioactive in-leakage. In addition to the annunciator, the component cooling water surge tank vent valves will automatically close. The sensitivity range of this monitor is based on Co-60.

Service Water Effluent Discharge Monitor

This channel continuously monitors the service water system (Section 14.2) discharge to the ultimate heat sink. An increase in activity would be indicative of radioactive in-leakage to this system. A gamma scintillation detector is used for this monitor, and its output is transmitted to the control room. The activity level of this system is indicated on a meter and a recorder. A high radiation level in this system will be annunciated in the control room. There is no automatic function associated with this channel. The sensitivity of this monitor is based on Co-60.

Waste Disposal System Liquid Effluent Monitor

This channel (Figure 16-4) continuously monitors all waste disposal system liquid releases from the waste monitor tanks (Section 15.1).

A Geiger-Mueller detector monitors all effluent discharges. The signal from this monitor is transmitted to the control room for indication on a meter and a recorder. A high radiation level on this discharge line will annunciate in the control room. In addition to the annunciator, the discharge valve located on the discharge line will automatically close. The discharge valve is located far enough downstream of the monitor to allow for the closure of this valve prior to any unplanned radioactive release. A single monitor is provided on each discharge line and is considered adequate since the monitor tanks are sampled and analyzed prior to any allowable discharge flow. The release of liquid waste is under administrative control, and this monitor is provided to maintain surveillance over the release.

Gas Decay Tank Effluent Gas Monitor

This channel (Figure 16-4) monitors the radioactivity released through the plant vent, especially during the venting of the gas decay tanks (Section 15.3). The detector is either a G-M tube or a beta scintillation detector. The detector output is transmitted to the control room for indication on a meter and a recorder. A high radioactivity condition will be annunciated in the control room. In addition to the annunciator, the isolation valve on the gas decay tank's vent line will automatically close. This will terminate the release and will initiate operator action to establish and correct the cause of this alarm. This monitor principally is used to detect Kr-85, Xe-133, and Xe-135, and its sensitivity is based on Kr-85.

Area Radiation Monitoring System

This system consists of channels which monitor and indicate the radiation levels in various physical areas of the plant. Table 16-2 lists the most common locations of area monitors. The detectors most generally used for area monitors are G-M tubes, beta or gamma scintillation detectors, and in some cases, air particulate with fixed filter collectors may be used. The detector output is transmitted to the control room where the radioactivity level is indicated and recorded. If the radiation level in a particular area exceeds a preselected setpoint, an annunciator will alarm in the control room.

The area radiation monitoring system will normally supply indication and alarms, both in the control room and locally, and will provide no automatic functions. However, there is one channel which will normally provide an automatic function. If the fuel handling building pool area monitor (Figure 16-4) should alarm, it will cause the fuel handling ventilation exhaust to be routed from its normal exhaust to a special exhaust system, comprised of booster fans and activated charcoal filters.

16.4 System Interrelations

16.4.1 Gross Failed Fuel Detector

There are several different methods used to detect failed fuel, of which only two methods will be explained below.

The first method is to monitor the radiation level in the chemical and volume control system volume control tank room. In the event of a failure of a fuel assembly or fuel element, the radioactive noble gas inventory in the volume control tank will increase, this results in a higher radiation level inside the volume control tank room, thereby alerting the operator to the failure of a fuel assembly. The expected or predicted radiation levels in the volume control tank room are as follows:

- Reactor Shutdown ~ 1 millirem per hour
- Reactor Operating ~ 100 millirem per hour
- 1% Failed Fuel ~ 1000 rem per hour

The problem with this type of system is that with increasing reactor power, a phenomenon occurs called iodine spiking. This iodine spiking will cause the radiation level in the volume control tank room to increase to high levels, giving this system a false indication.

A second method used to detect failed fuel is with a neutron detector, as shown in Figure 16-3. This system continuously monitors the reactor coolant system via a sample line from the hot legs, through two containment isolation valves, a sample cooler, neutron detector, then through a flow control device where the sample is discharged to the chemical and volume control system letdown line.

The sample is supplied to the neutron detector (BF3 proportional counter) via two containment isolation valves. These valves will automatically isolate on a Phase B isolation signal. The detector is sampling for delayed neutrons from short lived fission products, namely bromine-87, iodine-137, and bromine-88.

The 40 second delay prior to the exit of containment is for N-16 gamma considerations, and the remaining 20 seconds is to ensure that the detector is sampling these particular fission products. The time delay is established with the length of the sample line and the flow control device. Normally, the flow rate through this system will be set at approximately one gallon per minute.

The detector is a calibrated BF3 proportional counter whose indication ranges from 10^1 counts per minute (cpm) to 10^6 cpm on a logarithmic scale. Generally, there are two alarms associated with this system. The high alarm is normally set at 2×10^4 cpm above a preselected level. When this alarm actuates, a chemistry sample is required of the reactor coolant system. This alarm is indicative of the possibility of some fuel damage.

The other alarm is the high-high alarm, which is set at 1×10^5 cpm above a preselected level. Upon the receipt of this alarm, an immediate sample of the reactor coolant system is required, and reactor power is to be reduced 25 percent. This alarm is an indication of excessive fuel defects or failures.

16.5 Summary

The radiation monitoring system is a system that will measure the radiation or activity levels in various process streams or areas. It will provide information to the operator in the control room with the use of indicating meters, recorders, and alarms, both audible and visual. The radiation monitoring system is comprised of two subsystems. These subsystems are known as process monitors and area monitors. Tables 16-1 and 16-2 list some of the processes or areas that are generally monitored. In addition to the detector locations, these tables will list the most commonly used detectors for that specific location and the automatic functions, if any, that this particular channel may provide. In addition to these monitors, the facilities will be provided with some type of system or component to detect a gross failure of the fuel.

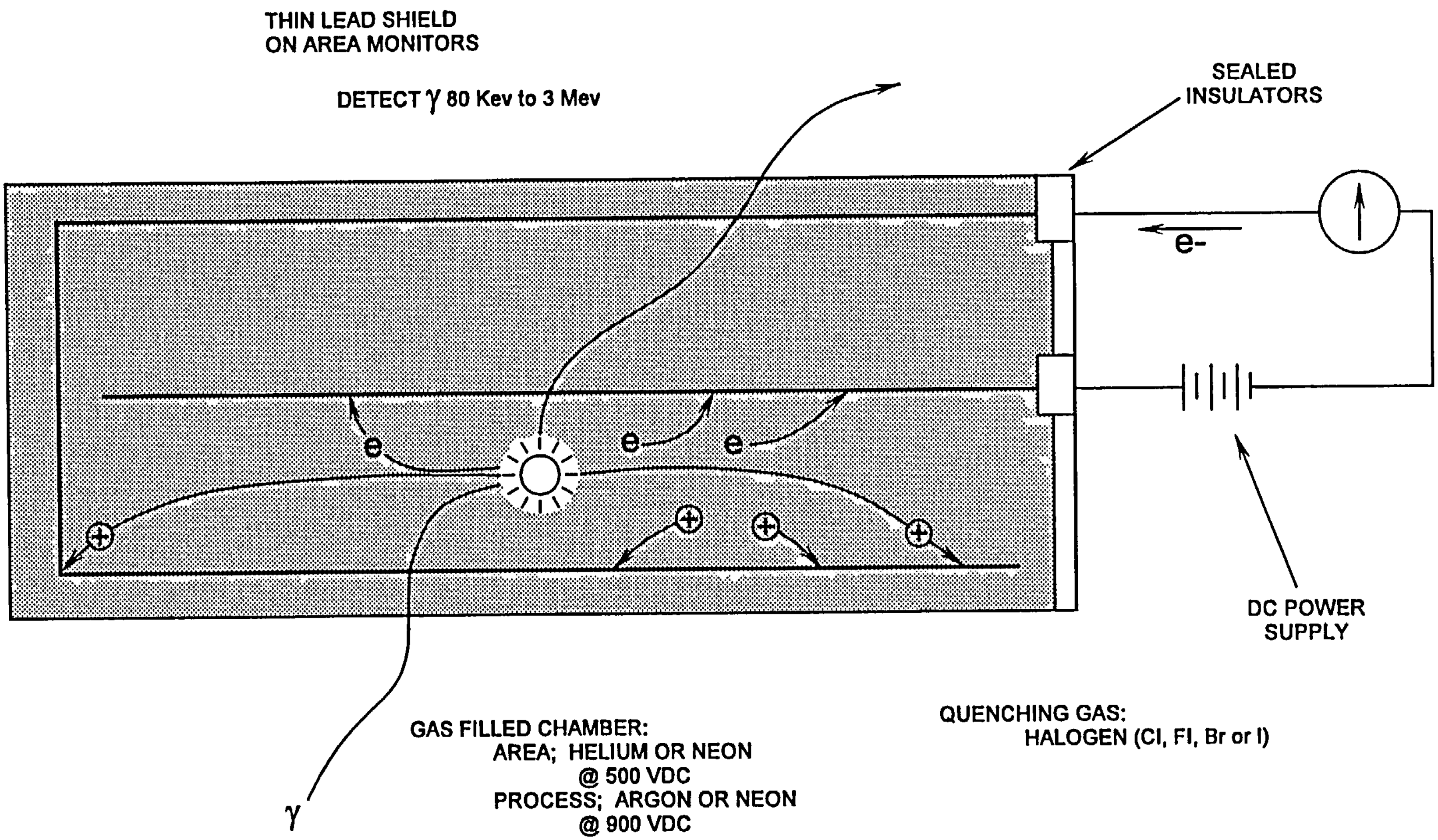

Figure 16-1 Geiger - Mueller Tube

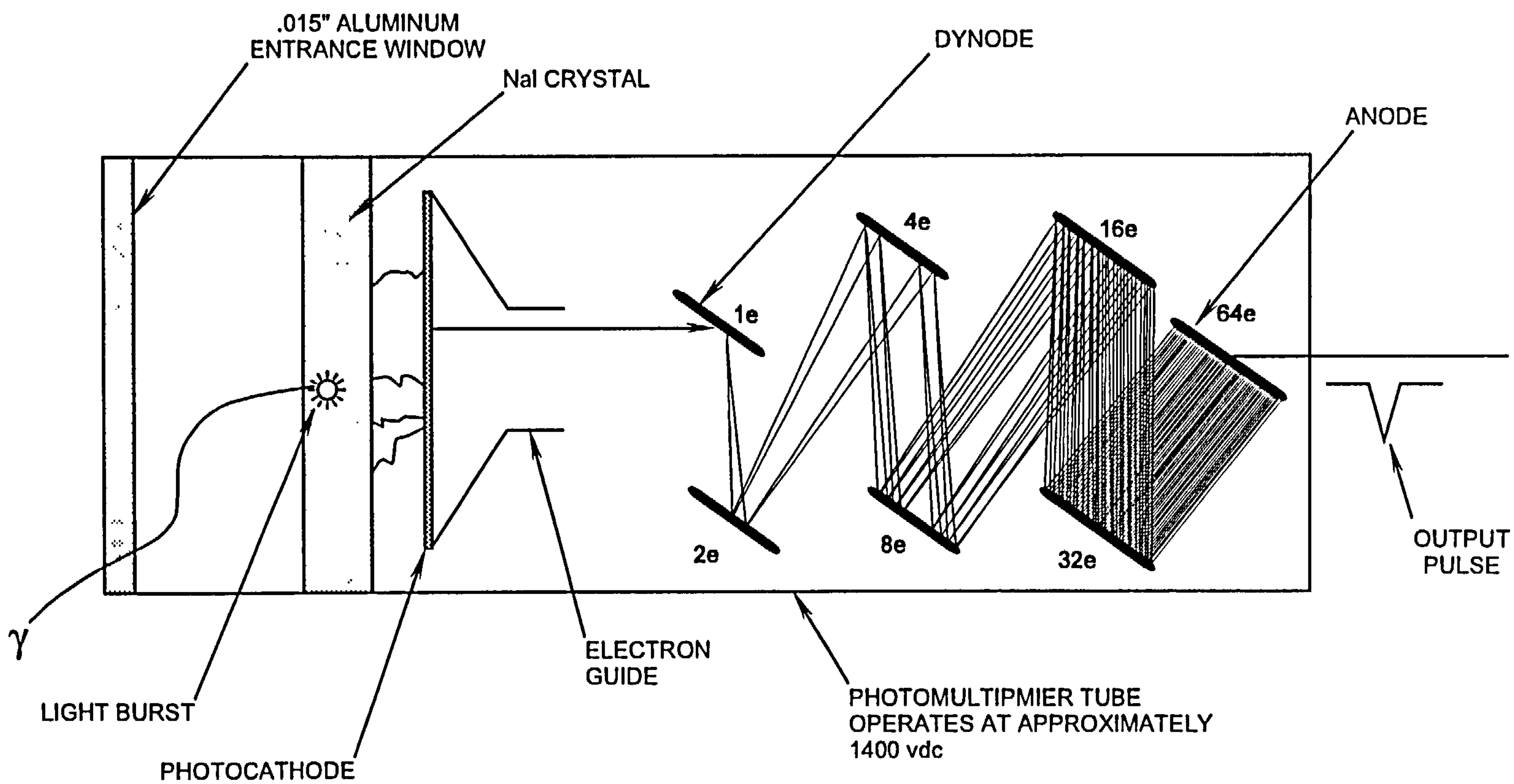

Figure 16-2 Scintillation Detector

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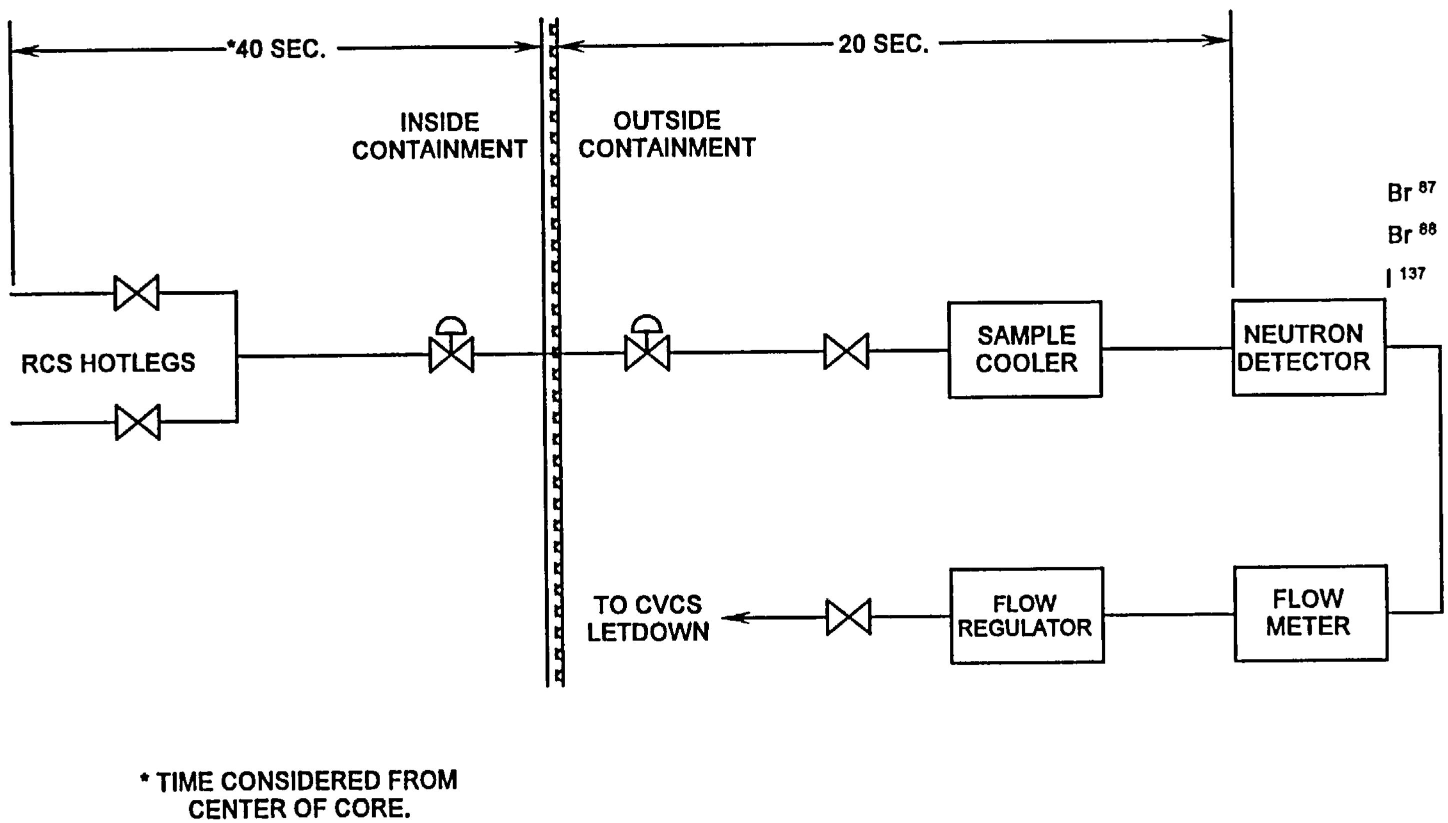

Figure 16-3 Gross Failed Fuel Detector

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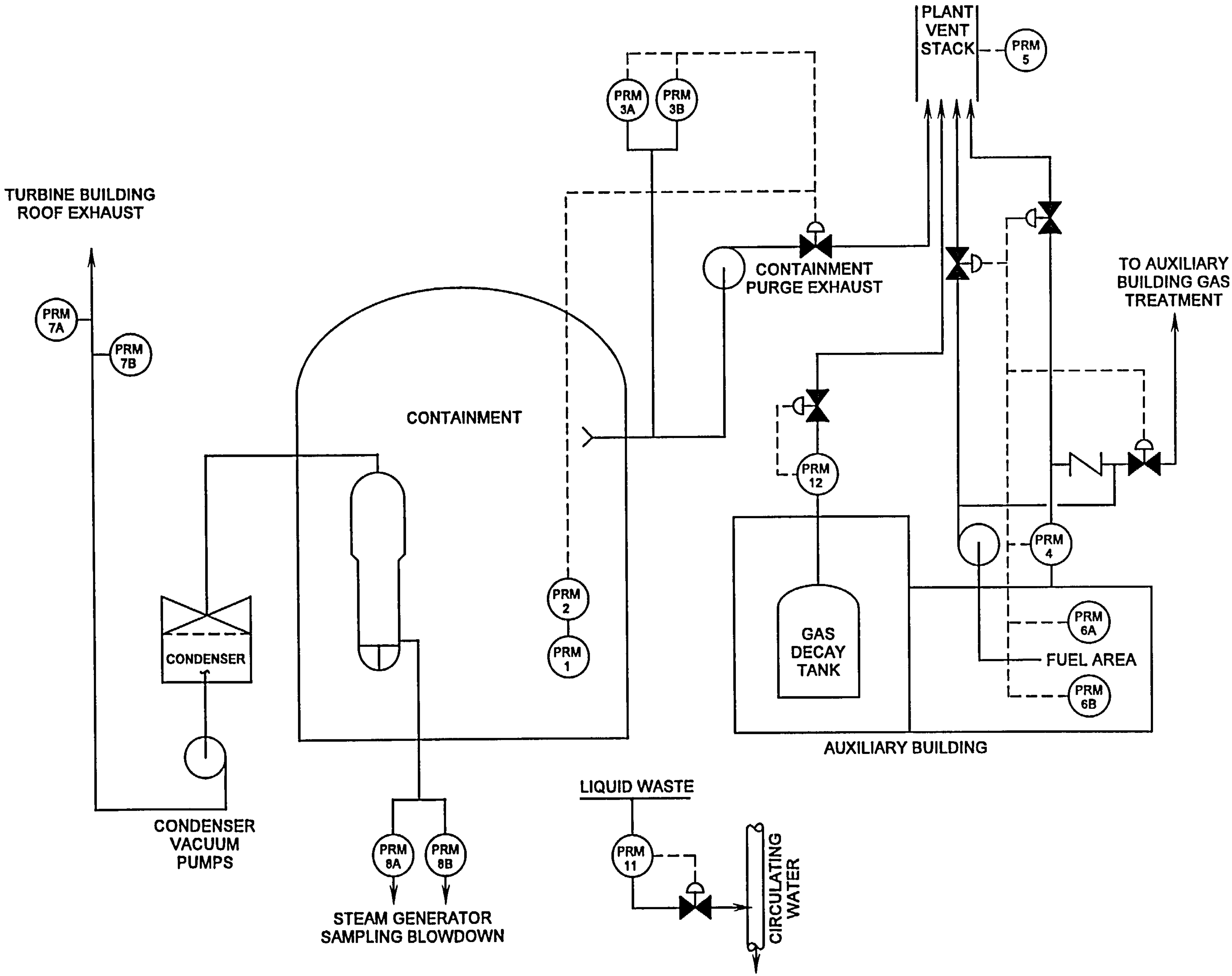

Figure 16-4 Radiation Monitor Locations

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Westinghouse Technology Systems Manual

Chapter 17

FUEL HANDLING AND STORAGE

Section

Westinghouse Technology Systems Manual

Section 17.1

Fuel Handling and Storage

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17.1 FUEL HANDLING AND STORAGE

Learning Objectives:

1. State the purposes of the fuel handling and storage systems.

2. State the functions of the following fuel handling system equipment:

   a. Spent fuel pool bridge crane
   b. New fuel elevator
   c. Fuel transfer canal
   d. Polar crane
   e. Manipulator crane
   f. RCCA change fixture
   g. Reactor vessel stud tensioner
   h. Conveyor car
   i. Upenders

3. State the reasons for handling spent fuel under water.

17.1.1 Introduction

The purposes of the fuel handling and storage systems are to receive, store, and transfer new and spent fuel. The fuel handling system has been designed to minimize the possibility of mishandling or operational errors that could cause fuel damage and result in a potential fission product release. The system is also designed to expedite the refueling operation while maintaining safe fuel handling practices. To facilitate this procedure, the reactor is refueled with equipment designed to handle the spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site.

This practice allows the operator to view the refueling operation while the water acts as a shield, moderator and coolant. The fuel handling system consists of three areas of operation which contain various systems and equipment. These areas are as follows:

1. Fuel Storage Building - where new and spent fuel is received, stored and shipped.
2. Containment - where the reactor is located and most of the handling of the fuel assemblies occurs during refueling operations.
3. Fuel Transfer System - which connects the fuel storage building with the containment and provides a means of transferring fuel assemblies between the two.

17.1.2 Fuel Storage Building

17.1.2.1 System Description

New fuel assemblies are received and stored in racks in the new fuel storage area (Figure 17.1-1). New fuel is delivered to the reactor by retrieving it from the new fuel storage area, transferring it to the new fuel elevator where it is lowered into the spent fuel pool, and taken through the transfer system into the containment. When inside the containment, the fuel is transferred to the reactor vessel where it is lowered into place. The transfer of fuel is done with the spent fuel pool, the fuel transfer canal, and the refueling cavity flooded for shielding requirements. The equipment used is designed to handle the spent fuel underwater.

The refueling operation is divided into five major phases:

1. preparation,
2. reactor disassembly,
3. fuel handling,
4. reactor assembly, and
5. spent fuel cask loading.

A general description of a typical refueling operation through the five phases is given below:

Phase I - Preparation

In order to minimize the radiation exposure during the refueling, the reactor coolant system is cleaned up and decontaminated as much as possible. This is accomplished by maximizing reactor coolant system cleanup using the chemical and volume control system prior to shutdown and after the reactor is shutdown. Hydrogen peroxide is added to remove deposited crud, which is then removed from the coolant by the chemical and volume control system. The containment atmosphere is cleaned up using the purge and filtration systems.

The reactor is shutdown and cooled to cold shutdown conditions with a final effective multiplication factor (K_{eff}) less than 0.95 (all rods in) and reactor coolant system temperature less than 140°F (MODE 6). Following a radiation survey, the containment structure is entered. At this time, the coolant level in the reactor vessel is lowered to a point slightly below the reactor vessel flange. Then the fuel transfer equipment and manipulator crane are checked for proper operation.

Phase II - Reactor Disassembly

The disassembly of the reactor vessel begins with the removal of the missile shield. All cables, air ducts, and insulation are then removed from the vessel head. The thermocouple conoseals are removed and protective covers are installed over the top of the thermocouple support columns. The incore instrumentation thimbles are disconnected from the transfer device, and the seals are broken at the seal table. The thimbles are then pulled back 24 feet through the bottom of the vessel. The reactor vessel stud tensioners are used to detension the vessel studs, and after removal of the studs, stud hole plugs are installed to protect the threads. Three stud holes are left unplugged and are fitted with threaded guide studs which help to position the various lifting rigs.

The refueling cavity is then prepared for flooding by sealing off the reactor cavity; checking the underwater lights, tools, and fuel transfer system; closing the refueling canal drain valves; and removing the blind flange from the fuel transfer tube. With the refueling cavity prepared for flooding, the vessel head is unseated and raised approximately 1 foot above the vessel flange. Water from the refueling water storage tank is pumped into the reactor coolant system by the residual heat removal pumps causing the water to overflow into the refueling cavity.

The vessel head and the water level in the refueling cavity are raised simultaneously, keeping the water level just below the head. When the water reaches a safe shielding depth, the vessel head is taken to its storage pedestal. The control rod drive shafts are then disconnected and, with the upper internals, are removed from the vessel. The fuel assemblies and rod cluster control assemblies are now free from obstructions and the core is ready for refueling.

Phase III - Fuel Handling

Fuel handling can start after the reactor has been subcritical for 100 hours (Technical Specification requirement). This allows time for the short-lived fission products to decay. The refueling sequence is started with the manipulator crane. As determined in the refueling guide which is prepared before each refueling, spent fuel assemblies are removed from the core. The positions of partially spent assemblies are changed (fuel shuffle) and new assemblies are added to the core.

The fuel handling sequence is:

1. The manipulator crane is positioned over a fuel assembly in the most depleted region of the core.
2. The fuel assembly is lifted by the manipulator crane to a predetermined height sufficient to clear the reactor vessel and still leave sufficient water covering to eliminate any radiation hazard to the operating personnel (minimum of 9.5 feet).
3. If the removed fuel assembly contains a rod cluster control assembly (RCCA), the fuel assembly will be placed in the RCCA change fixture by the manipulator crane. The rod cluster control assembly is then removed from the spent fuel assembly and placed into a new or partially spent fuel assembly also located in the change fixture.
4. The fuel transfer (conveyer) car is moved into position in the transfer canal in the containment near the upender.
5. The fuel assembly container section is pivoted to the vertical position by the upender.
6. The manipulator crane is moved to line up the fuel assembly with the fuel transfer system.
7. The manipulator crane loads a fuel assembly into the container section of the conveyor car.
8. The container is pivoted to the horizontal position by the upender.
9. The fuel assembly is moved through the fuel transfer tube to the spent fuel pool by the conveyor car.
10. The fuel assembly container is raised (see 17.2.3) to the vertical position. The fuel assembly is unloaded by the spent fuel pool bridge crane.
11. The fuel assembly is placed in the spent fuel storage rack.
12. A new fuel assembly is brought from dry storage, lowered into the transfer canal by the new fuel elevator, and loaded into the fuel assembly container of the conveyer car by the spent fuel pool bridge crane.
13. The fuel assembly container is pivoted to the horizontal position and the conveyor car is moved back into the containment.
14. Partially spent fuel assemblies are relocated (shuffled) in the reactor core, and new fuel assemblies are added to the core.
15. Any new fuel assembly or shuffled fuel assembly that will be placed in a control position is first placed in the rod cluster control assembly change fixture to receive a rod cluster control assembly from a spent fuel assembly.
16. This procedure is continued until refueling is completed.

Phase IV - Reactor Assembly

Reactor assembly, following refueling, is essentially achieved by reversing the operations given in Phase II - Reactor Disassembly.

Phase V - Spent Fuel Cask Loading

Spent fuel cask handling is as follows:

1. The fuel cask shipping conveyance is parked inside the fuel storage building.
2. The outside door is closed.
3. The shipping cask is picked up by the fuel storage building crane and is moved to an open area on the operating floor. If it is necessary to disengage the crane hook to free the crane for those uses, the cask is lowered to the cask decontamination facility or into the cask loading area of the spent fuel pool. In either of these locations, a seismic event would not overturn the cask.
4. The gate is placed in the slot between the spent fuel pool and the cask loading area.
5. The cask is picked up by the crane and is lowered onto the shelf in the loading area. The crane hook is disengaged from the cask, and an extension link is inserted between hook and cask. The cask then is lowered into the

deep portion of the pit.

6. The cask lid is removed and placed in the cask set down area.
7. The gate is removed from the slot.
8. Using the spent fuel bridge and hoist, fuel assemblies are transferred, one at a time, from the spent fuel storage racks to the cask.
9. The gate is placed in the slot, and the cask lid is replaced.
10. The cask is lifted onto the shelf, the extension link is removed, and the cask is removed from the loading area. It is then placed in the cask decontamination room for wash down.
11. The cask is moved to the cask set down area, and tie down devices are affixed while the cask undergoes preshipment tests.
12. The cask is placed on the shipping conveyance with the fuel storage building outer door closed.
13. The conveyance is then moved out of the building.

17.1.2.2 Component Description

The components of the fuel storage building will be discussed individually. The functions of each of the components will be described as the "component's overall function" or role in the fuel handling system.

New Fuel Storage Area

The new fuel is stored in vertical storage racks in the new fuel storage area (Figure 17.1-2). This area is constructed of reinforced concrete designed in accordance with the Seismic Category I requirements, except that the storage racks are not seismically qualified since there is no radiological hazard associated with their failure.

The new fuel storage area is sized for storage of fuel assemblies associated with the replacement of one-third of the core (76 assemblies). The center-to-center spacing (21 inches) is sufficient to ensure a $K_{eff} \leq 0.95$ assuming the storage area is flooded with unborated water.

Spent Fuel Pool

The construction of the spent fuel pool and storage area is of reinforced concrete with a seam welded, 1/4 inch thick, stainless steel liner (Figure 17.1-2). The spent fuel pool structure includes a storage area, a spent fuel cask loading area and a transfer canal which is connected with the refueling cavity in the containment. The pool is designed to accommodate 1,408 fuel assemblies (approximately 7.3 cores). The normal depth of borated water in the spent fuel storage pool is approximately 39 feet. The requirement for a water shield of 23 feet (Technical Specifications requirement) above the fuel assemblies must always be met. A high/low level alarm is annunciated in the control room when the water level rises to 1.5 inches above normal level or drops to 6 inches below normal level. The hoist on the spent fuel pool bridge crane has a mechanical stop which prohibits lifting a fuel assembly higher than a level at which its top is 9.5 feet (radiation shielding requirement) below the low level alarm point.

Located at one end of the spent fuel pool is the fuel transfer canal. This area is separated from the pool and contains an upender (fuel lifting mechanism), the transfer tube gate valve and rails for the transfer canal, a gate has been provided for isolation of one area from the other. With the gate closed the refueling cavity and fuel transfer canal can be drained and maintenance performed. The gate swings on hinges and is kept closed by a series of wedges along its length. Leak tightness is provided by a rubber seal.

The spent fuel storage racks are designed with sufficient center-to-center spacing (10.5 inches) between the assemblies and a fixed neutron absorber in the fuel cell walls to maintain K_{eff} less than 0.95 even if unborated water were used to fill the pool.

Fuel Storage Building Crane

An overhead crane has been provided for handling equipment in the fuel storage building. The crane will normally be used for the following operations:

1. Unloading of new fuel shipping containers.
2. Movement of fuel assemblies from the shipping containers to the new fuel storage area and from new fuel storage area to the spent fuel pool and the new fuel elevator.
3. Transfer of spent fuel casks.
4. Loading of spent fuel casks.

The crane is restricted from moving heavy loads over the spent fuel pool, the spent fuel cooling system, and engineered safety feature systems by electrical limit switches, which will de-energize the bridge drive, and mechanical stops which are installed on the rails. The purpose of the interlocks and stops is to protect the equipment from the damage that could be caused by a dropped load.

Spent Fuel Pool Bridge Crane

The spent fuel pool bridge crane is a wheel-mounted walkway, spanning the spent fuel pool and fuel transfer canal, Figure 17.1-3, which carries an electric monorail hoist on an overhead structure. The fuel assemblies are moved within the spent fuel pool and to and from the fuel transfer upender by means of the spent fuel handling tool. The tool's length is designed to limit the maximum lift of a fuel assembly to ensure a safe shielding depth and protect against lowering the hook into the pool water. Pointers located on the bridge and hoist when aligned with index marks on the side of the pool and bridge crane monorail provide the operator with proper location information of the various storage positions.

New Fuel Elevator

The transferring of new fuel from the new fuel storage area to the spent fuel pool is accomplished by the use of the fuel storage building crane and the new fuel elevator Figure 17.1-4. The new fuel elevator lowers the assembly from the surface of the spent fuel pool to the level of the spent fuel racks. There it can be latched by the spent fuel handling tool and transported to fuel transfer canal by the spent fuel pool bridge crane. This design eliminates lowering the fuel storage building crane hook and the new fuel handling tool into the spent fuel pool borated water.

New Fuel Assembly Handling Tool

This tool, commonly called the short-handled tool (25 inches long and weights 75 pounds), is used to handle new fuel on the operating deck of the fuel storage building (Figures 17-5 and 17-6), to remove the new fuel from the shipping container, to facilitate inspection and storage of the new fuel and loading of new fuel into the new fuel elevator. These operations are performed with the new fuel handling tool attached to the hook of the fuel storage building crane.

Spent Fuel Handling Tool

This tool, Figure 17.1-5, is used to handle new and spent fuel in the spent fuel pool. It is a manually actuated tool on the end of a long pole suspended from the spent fuel pool bridge crane. An operator on the spent fuel pool bridge guides and operates the tool.

17.1.2.3 New Fuel Receipt and Storage

New fuel arrives on site in metal shipping containers, Figure 17.1-7. These containers hold two fuel assemblies, with rod cluster control assemblies or burnable poison rod assemblies inserted, if needed. The total weight, with fuel assemblies, is approximately 4800 pounds. The shipping container is pressurized to 5 psig with air.

Several checks are made during fuel receipt to verify the shipping containers were not mishandled during shipment, mishandling could cause damage to the fuel assemblies. The first check made is to verify the container is pressurized. This is accomplished by opening a relief valve located at the end of the container, Figure 17.1-8.

The top of the container is then removed. The fuel assemblies are supported on a shock absorber mounted platform which is attached to the sides of the container. This platform must be raised to the vertical position for fuel assembly removal. Prior to raising the platform, a second check for mishandling is made by inspecting the overload indicators, Figure 17.1-9, which are mounted on the platform. Any excessive lateral motion will bend the indicators. Also located on the platform are accelerometer balls. These are steel balls held in position by spring pressure; any sudden jolt received by the container will cause the balls to move from under the springs. After all checks have been completed, the platform is raised to the vertical position using the fuel storage building crane.

Next, the new fuel handling tool is connected to the fuel storage building crane and then attached to the new fuel assembly. The fuel assembly is then removed from the platform. The fuel assembly is transported, in the vertical position using the fuel storage building crane, to the new fuel storage area. A visual inspection is made as a last check for damage and the new fuel assembly is lowered into its storage in the new fuel storage racks.

17.1.3 Containment

17.1.3.1 System Description

The containment building, Figures 17.1-1 and 17.1-10, contains the reactor, the refueling cavity, equipment storage areas and part of the fuel transfer system. In the refueling cavity, fuel is removed from the reactor vessel, and transferred by a manipulator crane to the fuel transfer system for movement into the fuel storage building.

17.1.3.2 Component Description

The components of the fuel handling system inside the containment are discussed individually. In each case the component description, purpose and function is provided.

Refueling Cavity

The refueling cavity, Figure 17.0-1 and 17.0-10, is a stainless steel lined, reinforced concrete structure that forms a pool above the reactor when it is filled with borated water during refueling. The borated water is pumped from the refueling water storage tank (RWST) into the cavity by the residual heat removal system. Radiation at the surface of the water is designed to be < 2.5 mr/hr during fuel assembly transfer. (At all times during transfer the irradiated fuel is handled underwater.)

Vessel Area of Refueling Cavity

Prior to refueling operations, and after reactor cooldown, the reactor vessel flange is sealed to the bottom of the reactor cavity, Figure 17.0-10, by a clamped, gasketed seal. This seal ring prevents leakage of refueling water from the cavity into ventilation wells and the area beneath the reactor vessel. Special covers also are installed to seal off the excore detector wells. The floor and sides of the reactor cavity are lined with stainless steel to insure against leakage and to prevent contact of the coolant with the reinforced concrete walls of the cavity.

Fuel Transfer Canal

The canal is formed by two concrete shield walls, which extend upward to the same elevation as the refueling cavity. The floor of the fuel transfer canal and a portion of the refueling cavity is at a lower elevation than the reactor flange to provide the greater depth required for operation of the fuel transfer system upenders and the rod cluster control assembly change fixture. The fuel transfer tube enters the reactor containment and protrudes through the end of the fuel transfer canal. The fuel transfer tube is a 20 inch stainless steel pipe which connects the fuel transfer canal in the containment with the fuel canal in the fuel storage building.

Polar Crane

A large overhead crane has been provided to handle equipment inside containment. It is used to lift the reactor vessel head, and the reactor internals during the refueling sequence.

Manipulator Crane

The manipulator crane, Figure 17.1-11, is used to remove, replace and position fuel assemblies within the core. The manipulator consists of a rectilinear bridge and trolley with a vertical mast which extends into the refueling water. The controls for the following components, which are located on the manipulator crane, are shown on Figure 17.1-12. The bridge, which spans the reactor cavity, runs on rails set into the operating deck of the containment along the edge of the refueling cavity and transfer canal. The trolley runs on the bridge and positions the operators platform and mast assembly across the width of the refueling cavity.

Gripper Mast Assembly

The gripper assembly, Figure 17.1-13, is mounted on the bottom of the gripper tube. The gripper tube telescopes into and out of the mast. A hoist on the manipulator crane trolley raises and lowers this gripper tube. Movement of the gripper tube within the mast is guided by 7 sets of 3 roller bearings. The 3 rollers in each set are spaced evenly at 120 degree intervals and prevent the gripper tube from hanging up or swinging freely in the mast. The gripper assembly is air operated with air pressure needed to disengage the fingers. Raising and lowering of the gripper tube and gripper assembly is accomplished by the gripper tube hoist. The gripper tube is long enough so that the upper end is still contained in the mast when the gripper assembly contacts the fuel, yet short enough so that when the fuel is raised it is entirely contained within the mast to provide protection for the fuel assembly while being transported in the refueling cavity. Thus, fuel assembly protection and some additional shielding from the fuel assembly is provided. The mast is normally held stationary but may be rotated 300 degrees manually by retracting a position stop button and turning the outer mast with a turning bar.

Safety and travel interlocks associated with the manipulator crane are listed in Table 17-1. The interlocks are designed to prevent damage to the fuel assembly being moved and the fuel remaining in the vessel. Figure 17.1-14 shows typical travel limits for the bridge and trolley positions. In an emergency, and for fine adjustments in position the bridge, trolley and winch can be operated manually using a hand wheel on the individual motor shafts.

Rod Cluster Control Assembly (RCCA) Change Fixture

The RCCA Change Fixture, Figure 17.1-15, is mounted on the transfer canal wall and is used in removing rod cluster control and spider mounted secondary source assemblies from spent fuel assemblies and inserting them into new or partially spent fuel assemblies. The fixture consists of two main components: a guide tube mounted to the wall for containing and guiding the withdrawn Rod Cluster Control (RCC) assemblies, and a mounted carriage for holding the fuel assemblies under the guide tube.

A rod cluster control assembly can be removed by the RCCA change fixture gripper and hoist. The rod cluster control assembly can be aligned with the other fuel assembly's guide tubes for insertion into the new fuel assembly. The RCCA change fixture gripper is raised and lowered in the guide tube by a cable driven from a hoist.

The RCCA change fixture gripper is pneumatically operated to latch and unlatch the RCCA assemblies. The wheel mounted carriage support is anchored to the floor of the refueling cavity. The carriage contains compartments for two fuel assemblies and one RCC element with each one capable of being positioned by a chain and cable assembly operated by a hand winch from the operating floor. This winch has a lock or shaft clamp on it to prevent movement. Two stationary stops have been attached to the extremes of the support frame. These stops will prevent the carriage from rolling off the ends of the tracks. Positioning stops are also provided on both the carriage and frame to locate each of the three carriage compartments directly below the guide tube.

Reactor Vessel Head Lifting Device

The reactor vessel head lifting device, Figure 17.1-16, consists of a welded and bolted structural steel frame with suitable rigging to enable the polar crane operator to lift the head and store it during refueling operations. The lifting device is permanently attached to the reactor vessel head. Attached to the head lifting device are the monorail and hoists for the reactor vessel stud tensioners.

Reactor Internals Lifting Device

The reactor internals lifting device, Figure 17.1-17, is a structural frame suspended from the polar crane. The frame is lowered onto the guide tube support plate of the internals, and is manually bolted to the support plate by three bolts. Bushings on the frame engage guide studs in the vessel flange to provide guidance during removal and replacement of the internals package. The reactor internals lifting device is used to lift the upper internals package as well as the lower reactor internals package.

Reactor Vessel Stud Tensioner

Stud tensioners, Figure 17.1-18, are employed to secure the head closure joint at every refueling. The stud tensioner is a hydraulically operated device that uses oil as the working fluid. Stud tensioners minimize the time required for the tensioning or unloading operations of the reactor vessel head bolts. Three tensioners are provided and are applied simultaneously to three studs located 120 degrees apart. A single hydraulic pumping unit operates the tensioners, which are hydraulically connected in series. The studs are tensioned to their operational load in two steps to prevent high stresses in the flange region and unequal loadings in the studs. Relief valves on each tensioner prevent overtensioning of the studs due to excessive pressure.

Special Refueling Tools

Control Rod Drive Shaft Unlatching Tool

The control rod drive shafts are unlatched and latched to the full length rod cluster assembly spiders using the control rod drive shaft unlatching tool, Figures 17.1-5 & 17.1-19. This tool is suspended from the auxiliary hoist on the manipulator crane and is operated from the bridge. The latching mechanism is pneumatically operated. All full length RCCA drive shafts are removed as a unit with the reactor vessel upper internals.

Burnable Poison Rod Assembly (BPRA) Handling Tool

The burnable poison rod assembly handling tool, Figure 17.1-5, is a long handled tool used in the spent fuel pool and fuel transfer canal to transfer irradiated burnable poison rod assemblies between two fuel assemblies or between a fuel assembly and a special insert temporarily placed on selected spent fuel assembly storage racks. The tool is suspended from the hoist on the spent fuel pool bridge. An operator standing on the bridge guides the tool and manually actuates the engagement and handling mechanisms.

Irradiation Sample Handling Tool

The irradiation sample handling tool, Figure 17.1-5, is a long handled tool used to remove the irradiation specimens from their holders located on the outer surface of either the thermal shield or the neutron pads inside the reactor vessel. The tool is suspended from the polar crane and operated from the manipulator crane bridge.

Rod Cluster Control Thimble Plug Tool

The rod cluster control thimble plug tool is a manually operated tool, Figure 17.1-5, and is used in the transfer canal to remove or insert the thimble plug in a fuel assembly. When transferring an RCCA from one fuel assembly to another, a thimble plug is inserted in the fuel assembly from which the RCCA was removed.

17.1.4 Fuel Transfer System

17.1.4.1 System Description

The fuel transfer system is located in the fuel transfer canal area of the containment and fuel storage building, and utilizes an underwater conveyor to move fuel between these two areas, Figure 17.1-10. The underwater air-motor driven conveyor system runs on tracks extending from the containment through the transfer tube and into the fuel storage building. The container section of the conveyor car receives a fuel assembly is then lowered by the upender to a horizontal position for passage through the tube. Following that operation it is raised to a vertical position by a second upender in the fuel transfer canal (in the spent fuel pool side). The spent fuel pool bridge hoist then removes the assembly from the conveyor and places it in storage. A blind flange supplied with containment penetration pressurization air is bolted on the transfer tube inside the containment to seal the reactor containment.

17.1.4.2 Component Description

Conveyor Car Assembly

The conveyor car assembly is made up of two parts: the conveyor car frame and the fuel assembly container. The conveyor car frame is built out of a long stainless steel pipe. Mounted on and welded to the pipe at eight locations are wheel assemblies. The wheels ride on tracks which extend from the containment to the fuel storage building. Located on the bottom of the car frame along its entire length is welded a roller chain. Two gears located on the drive frame assembly engage the chain and provide the driving force.

The fuel assembly container is pinioned at one end to the conveyor car and is capable of being rotated to the vertical position about this point. The container is provided with locating guides which mate with pins on the upender. It is through the pins that the upender attaches itself to the fuel assembly container. During normal operation the car travels without any difficulty; if the car were to become stuck in the tube or the transfer canal, the car can be retrieved by pulling on its attached emergency cable with the fuel storage building crane.

Drive Frame Assembly

The drive frame assembly consists of a two speed reversible air driven motor which turns two gears that are connected to the motor by a roller chain. The two gears engage the chain welded to the bottom of the conveyor car frame. By rotating the gears with the drive motor, the car is propelled along the track. The air to the drive motor is turned on and off through the operation of solenoid valves. The solenoid valves are controlled from the reactor side control panel.

Upenders (Lifting Mechanisms)

The upenders are made up of an "I" beam that pivots about a support which mates with guides located on the fuel assembly container support structure. The upender is raised and lowered with a cable driven by an electrically operated winch. A hand wheel has been provided to manually operate the winch.

Gate Valve

A wedge type gate valve is installed at the fuel storage building side of the fuel transfer tube to provide a means of isolating the fuel transfer tube. The valve is large enough to allow the conveyor car to pass freely.

Fuel Transfer Control System

The fuel transfer control system is located on two panels. The conveyor car and reactor side upender are operated from the panel located inside containment while the fuel storage building upender is controlled from the panel located in the fuel storage building. Various procedural requirements and an interlock system between the two control points provide adequate fuel, equipment, and personnel protection during the operation of the fuel transfer system.

17.1.5 Summary

The maximum design stress for the structures and for all parts involved in gripping, supporting, or hoisting the fuel assemblies is 1/5 of the ultimate strength of the material. This requirement applies to normal working load and emergency pullout loads, when specified, but not the earthquake loading. To resist safe shutdown earthquake forces, the equipment is designed to limit the stress for a combination of normal working forces plus safe shutdown earthquake forces.

The fuel handling building crane is provided to move new fuel and spent fuel casks in the fuel handling building. Movement of these loads by the fuel building crane is allowed in all areas of crane travel except directly over the spent fuel storage racks. These interlocks (mechanical stops) will help to eliminate the possibility of accidentally damaging the spent fuel.

The fuel transfer tube connecting the fuel transfer canal inside the containment and the fuel transfer canal in the fuel storage building is closed on the containment side by a blind flange at all times except during refueling operations. Two seals are located around the periphery of the blind flange with leak check provisions between them. The fuel transfer tube is isolated on the fuel storage buildings side by the gate valve.

During all phases of spent fuel transfer, the gamma dose rate, 3 feet above the surface of the water, is < 2.5 mr/hr or less. This is accomplished by maintaining a minimum of 9.5 feet of water above the top of an active fuel assembly during all handling operations.

The two cranes used to lift spent fuel assemblies are the manipulator crane and the spent fuel pool bridge hoist. The manipulator crane contains positive stops which prevent the top of a fuel assembly from being raised to within of 10 feet the normal water level in the refueling cavity. The hoist on the spent fuel pool bridge crane moves spent fuel assemblies with a long handled tool. Hoist travel and tool length limit the maximum lift of a assembly in the spent fuel pool.

As part of normal plant operations, the fuel handling equipment is inspected for operating conditions prior to each refueling operations. During the operational testing of this equipment, procedures are followed that will verify the correct performance of the fuel handling system interlocks.

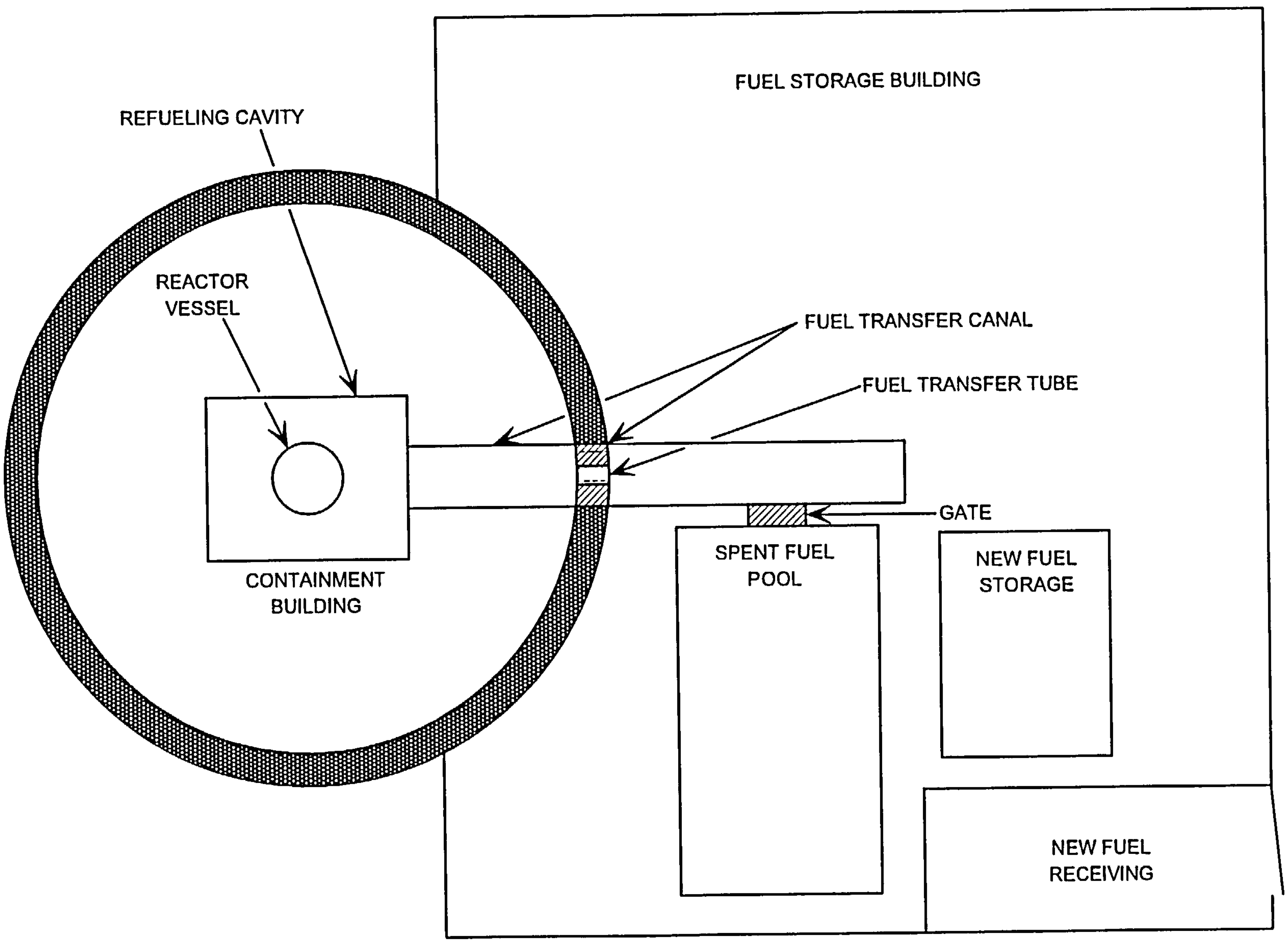

Figure 17.1-1 Containment - Fuel Handling Area Layout

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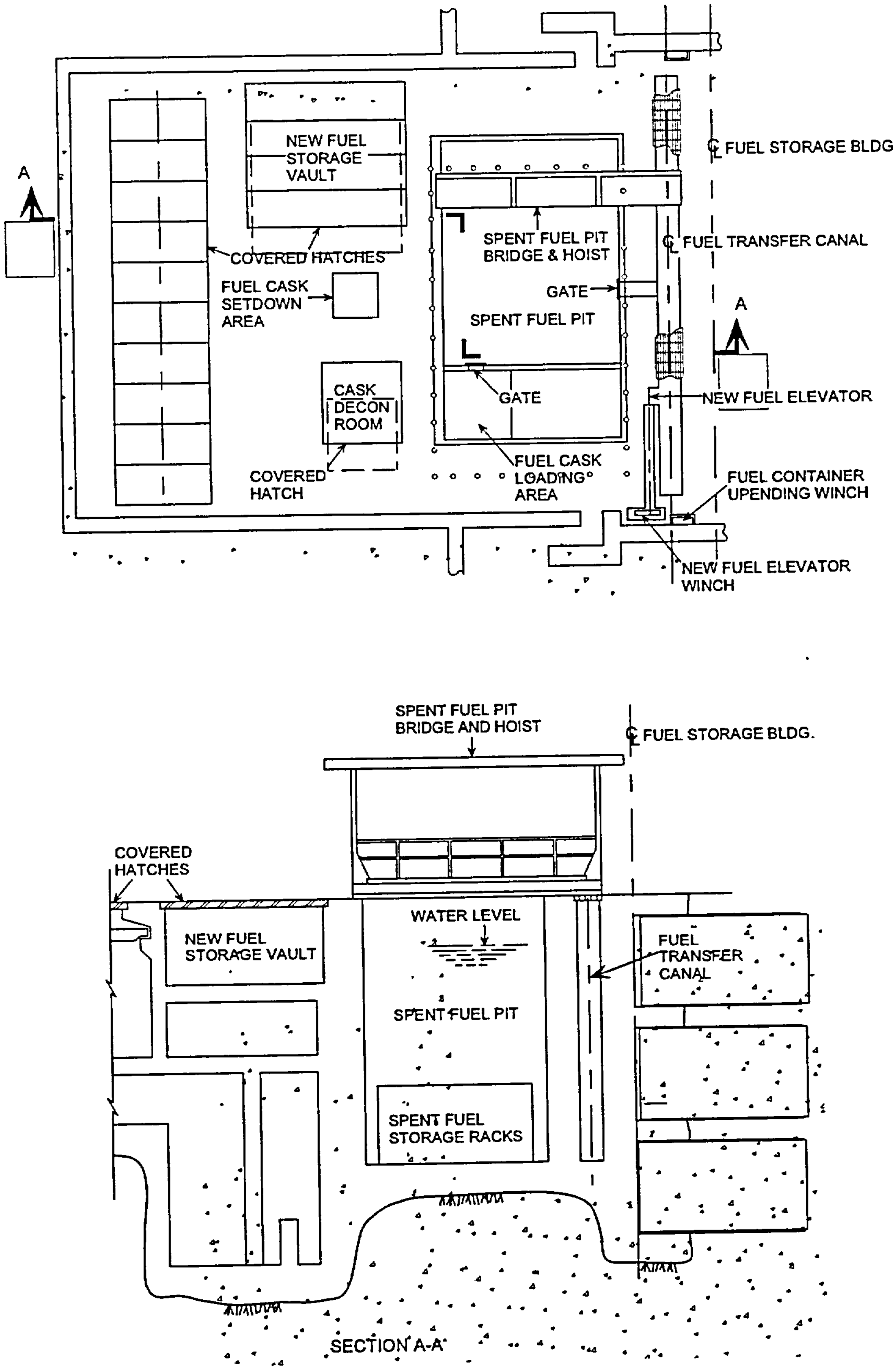
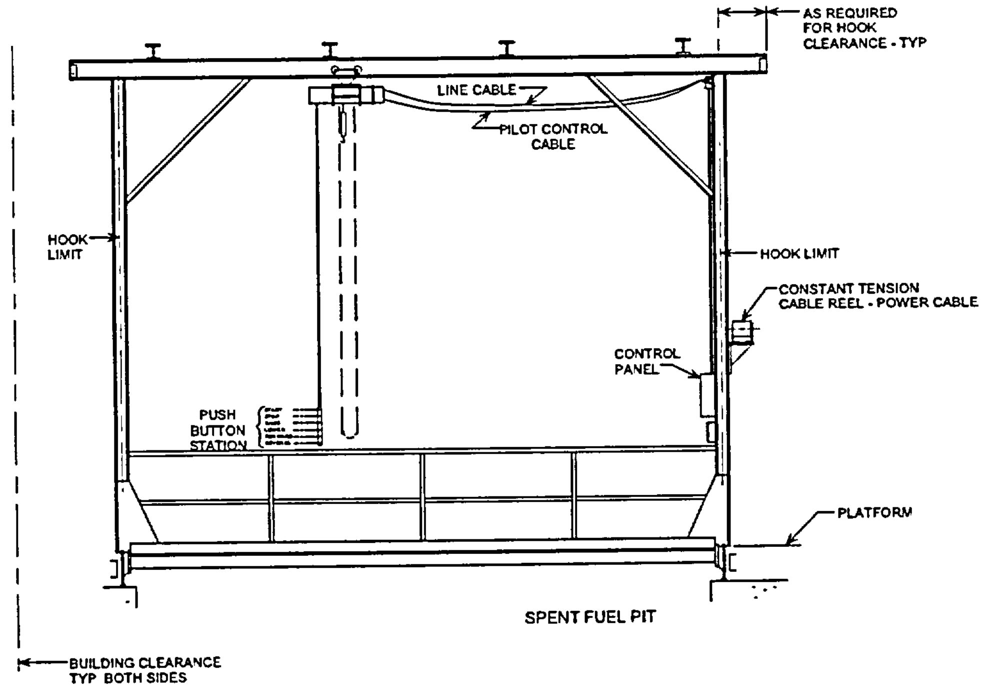


Figure 17.1-2 Fuel Storage and Handling



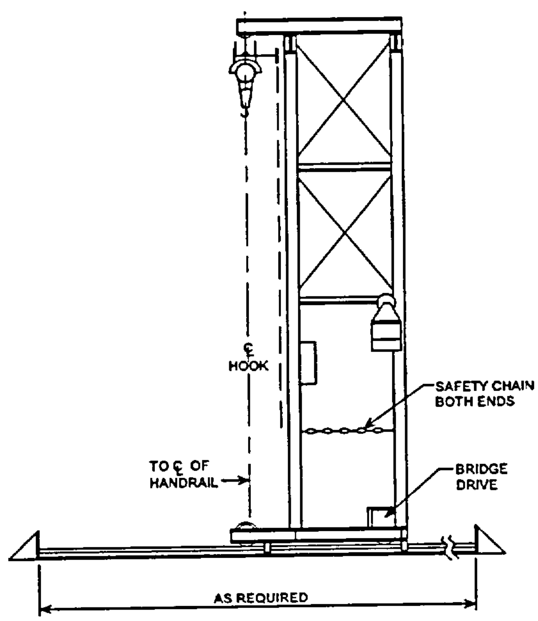


Figure 17.1-3 Spent Fuel Pit Bridge

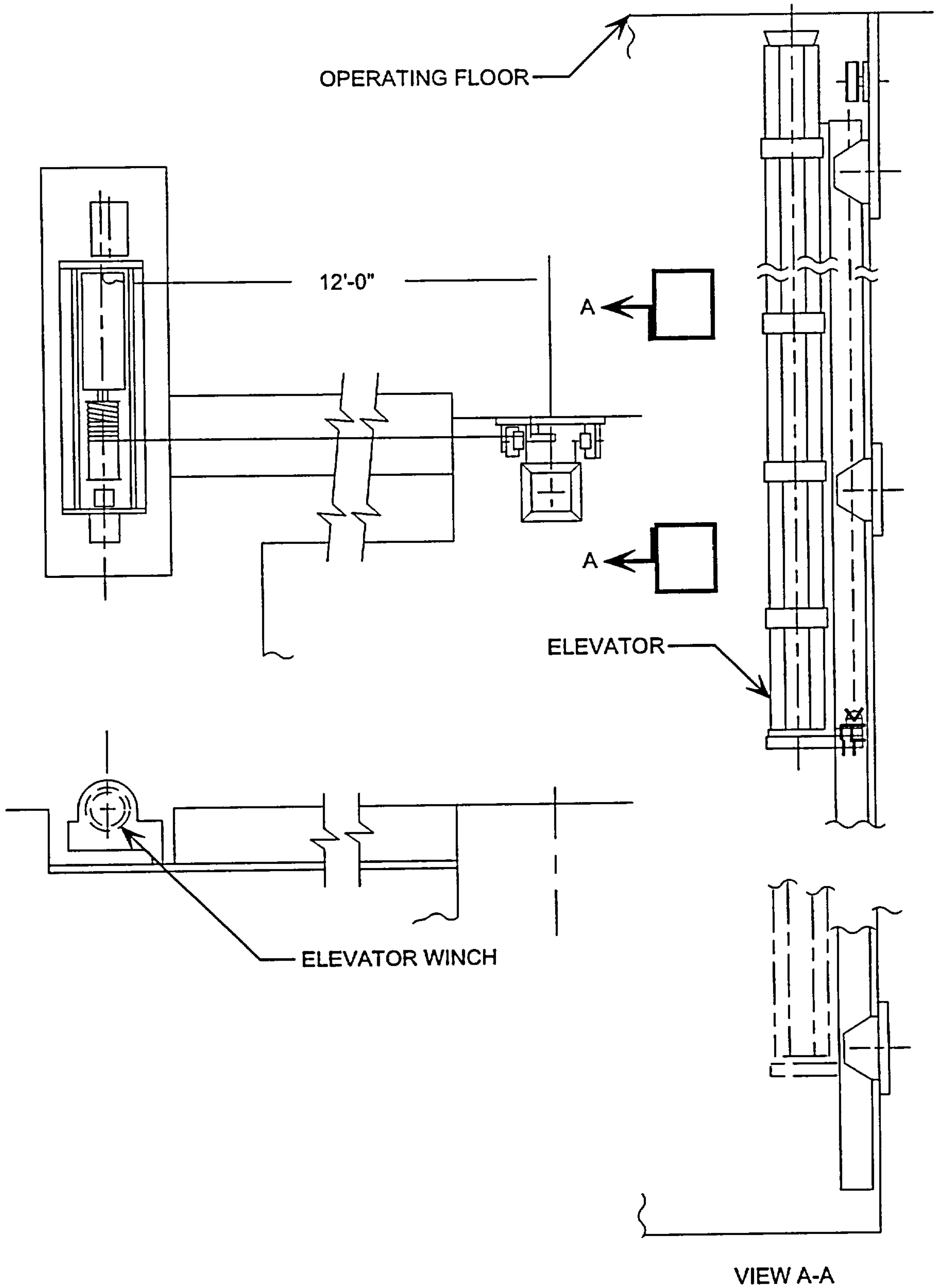


Figure 17.1-4 New Fuel Elevator

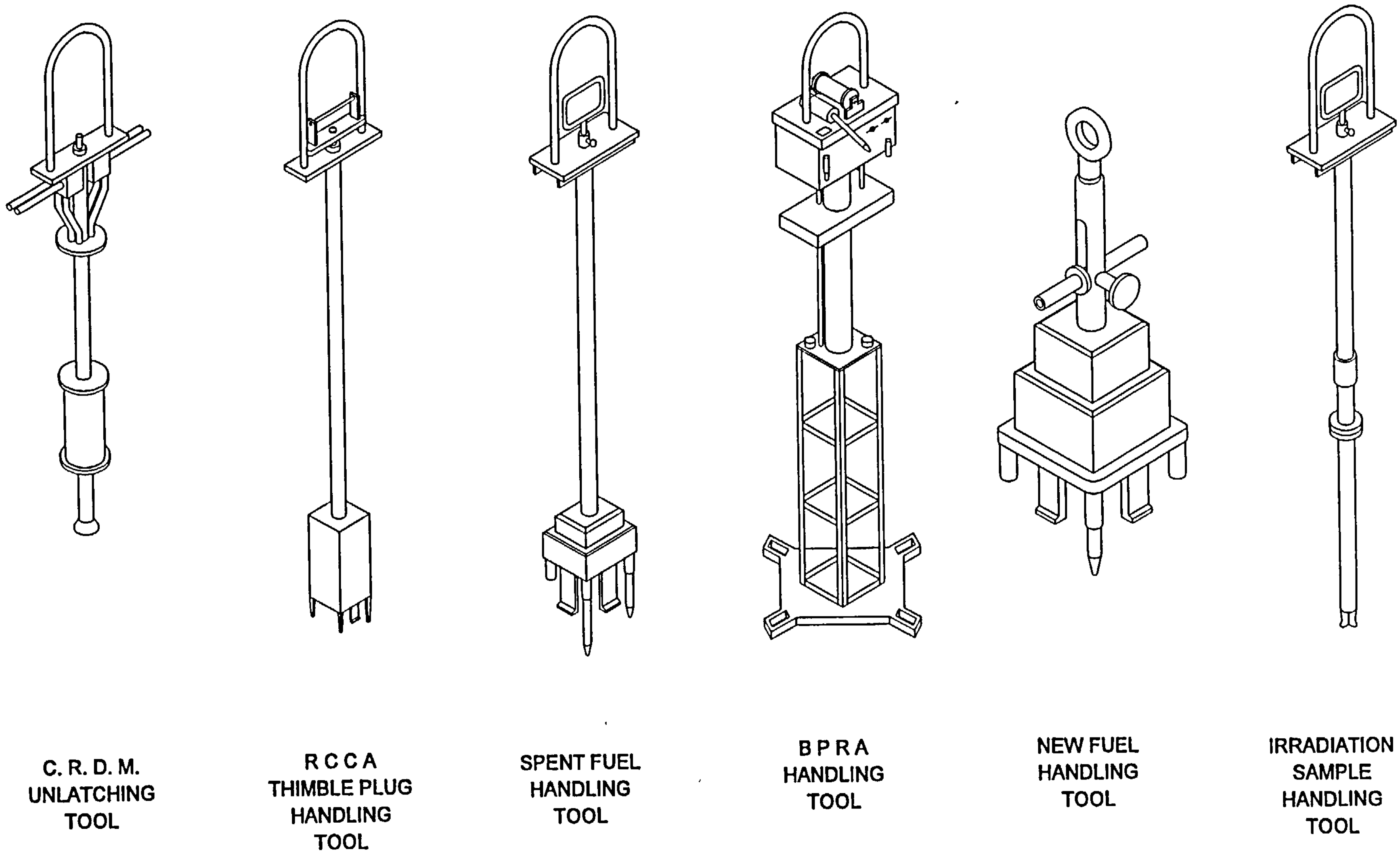


Figure 17.1-5 Fuel Handling Tools

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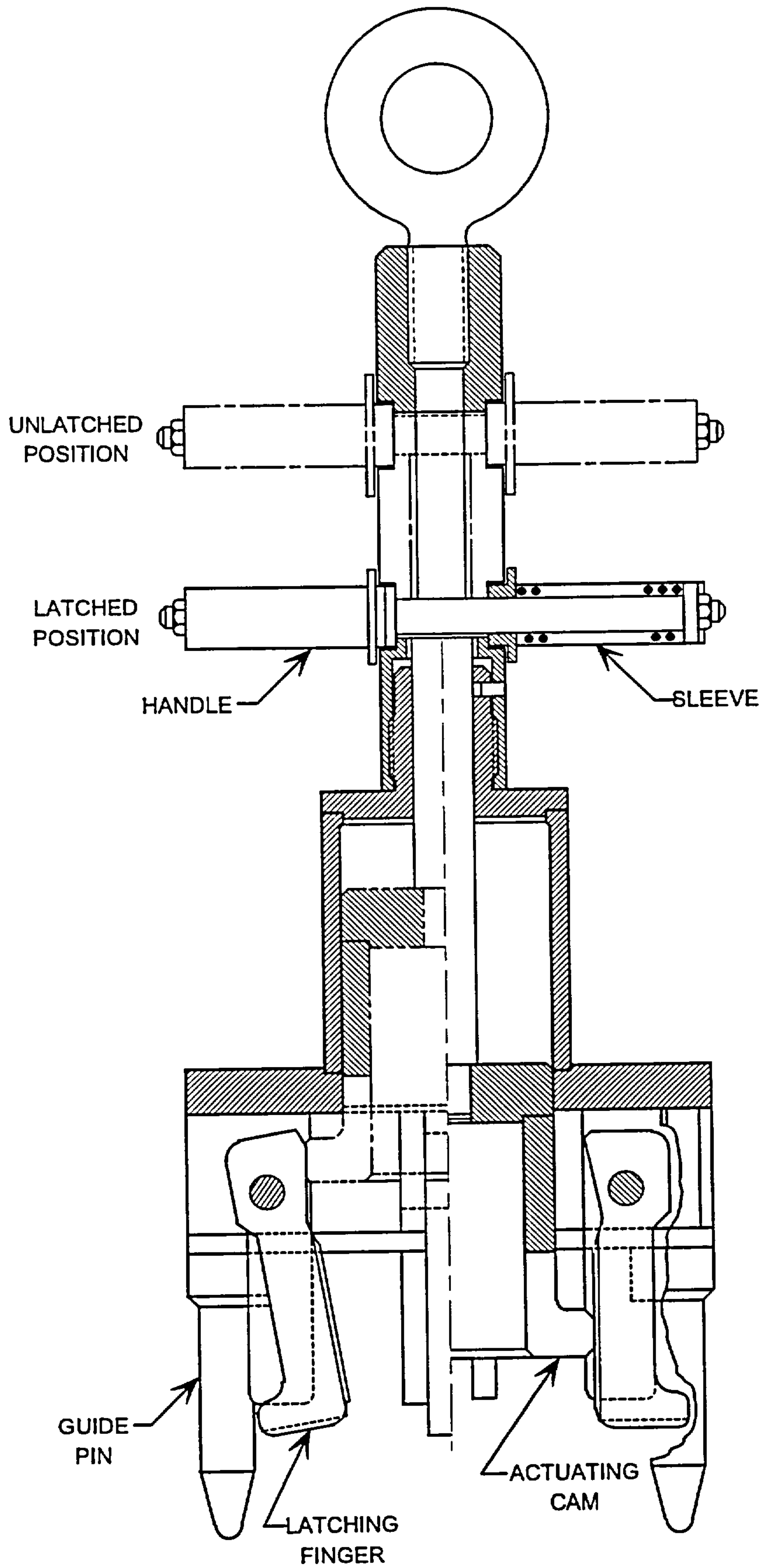

Figure 17.1-6 New Fuel Handling Tool

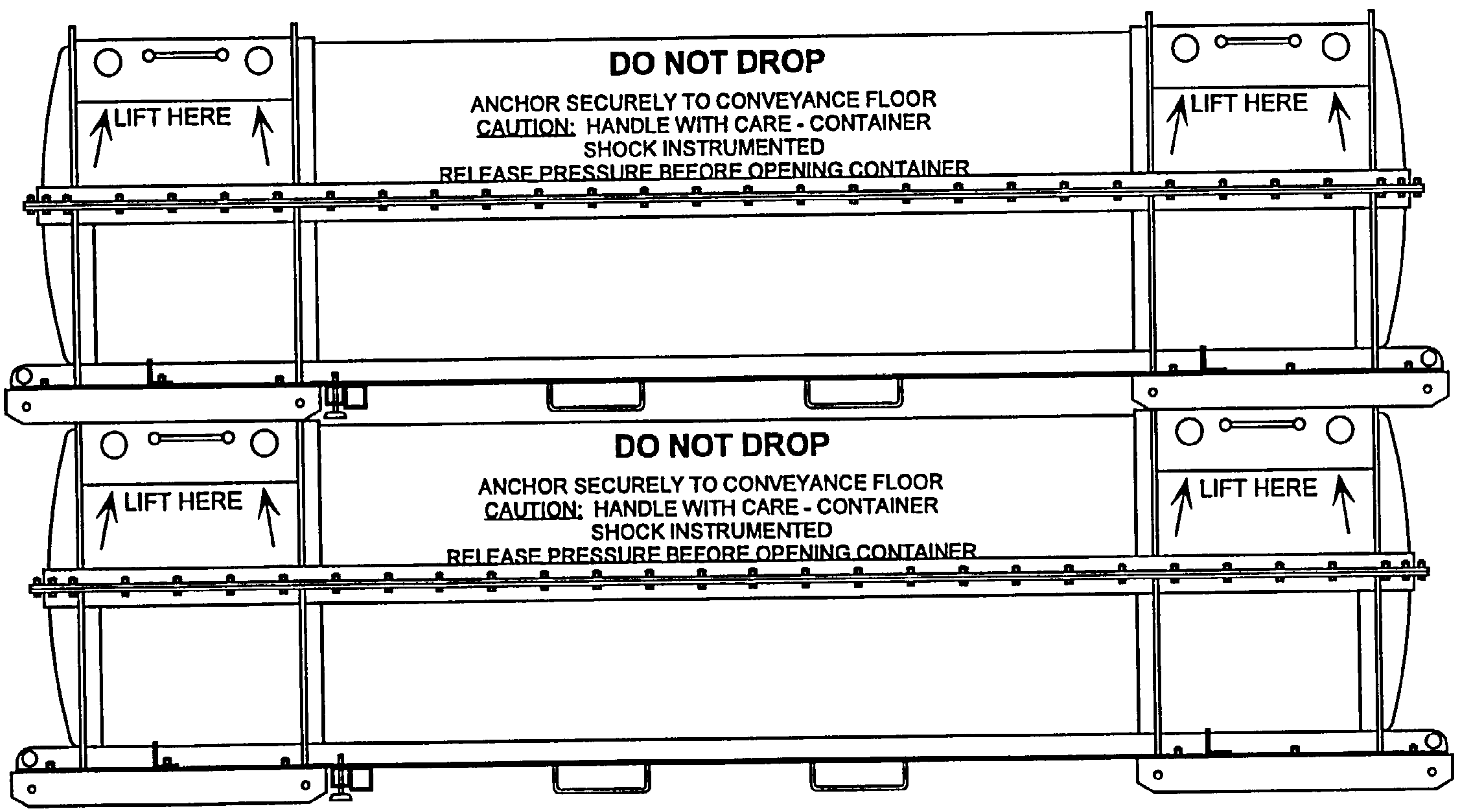

Figure 17.1-7 Loaded Shipping Container

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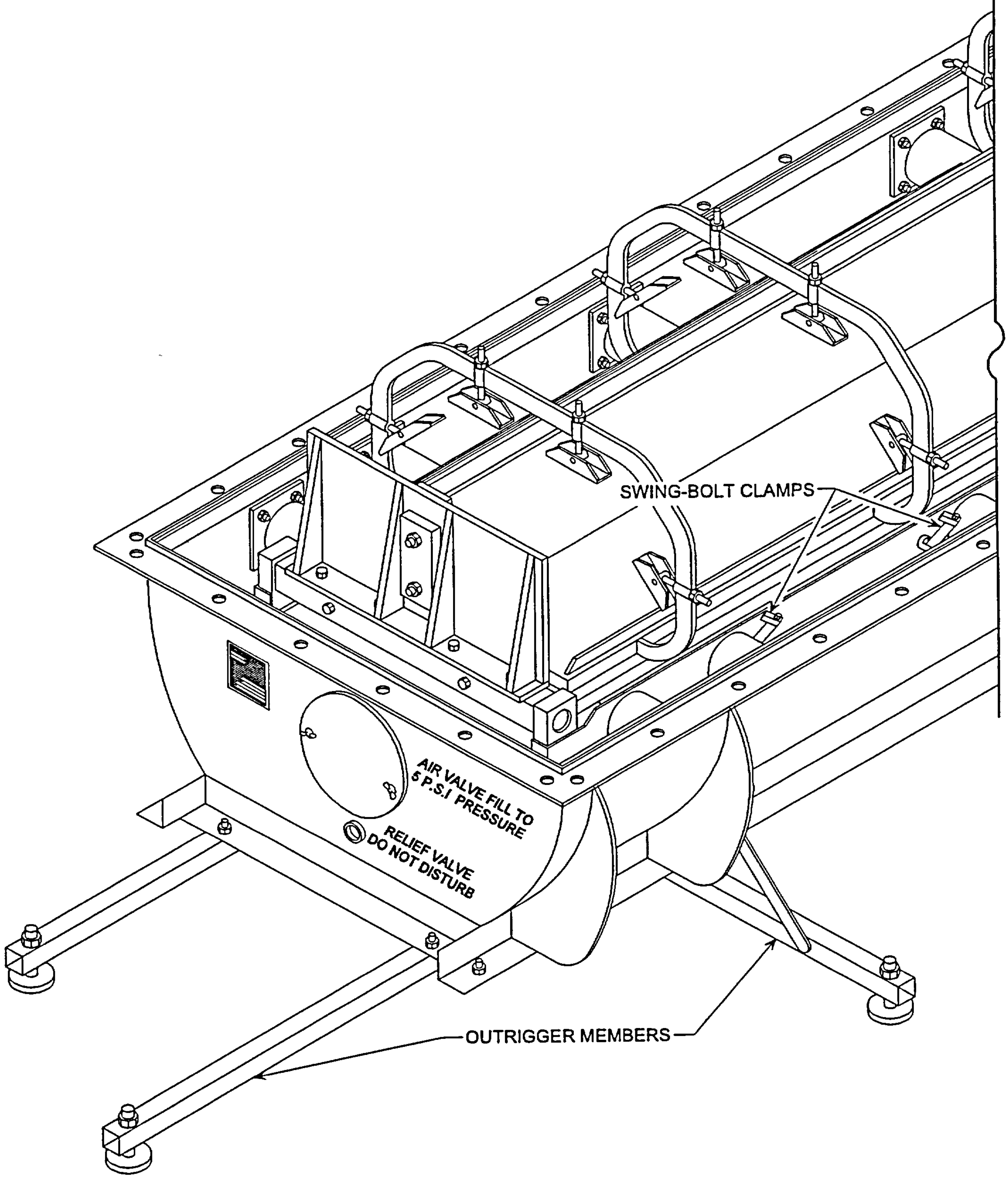

Figure 17.1-8 Preparation for Uprighting Internals

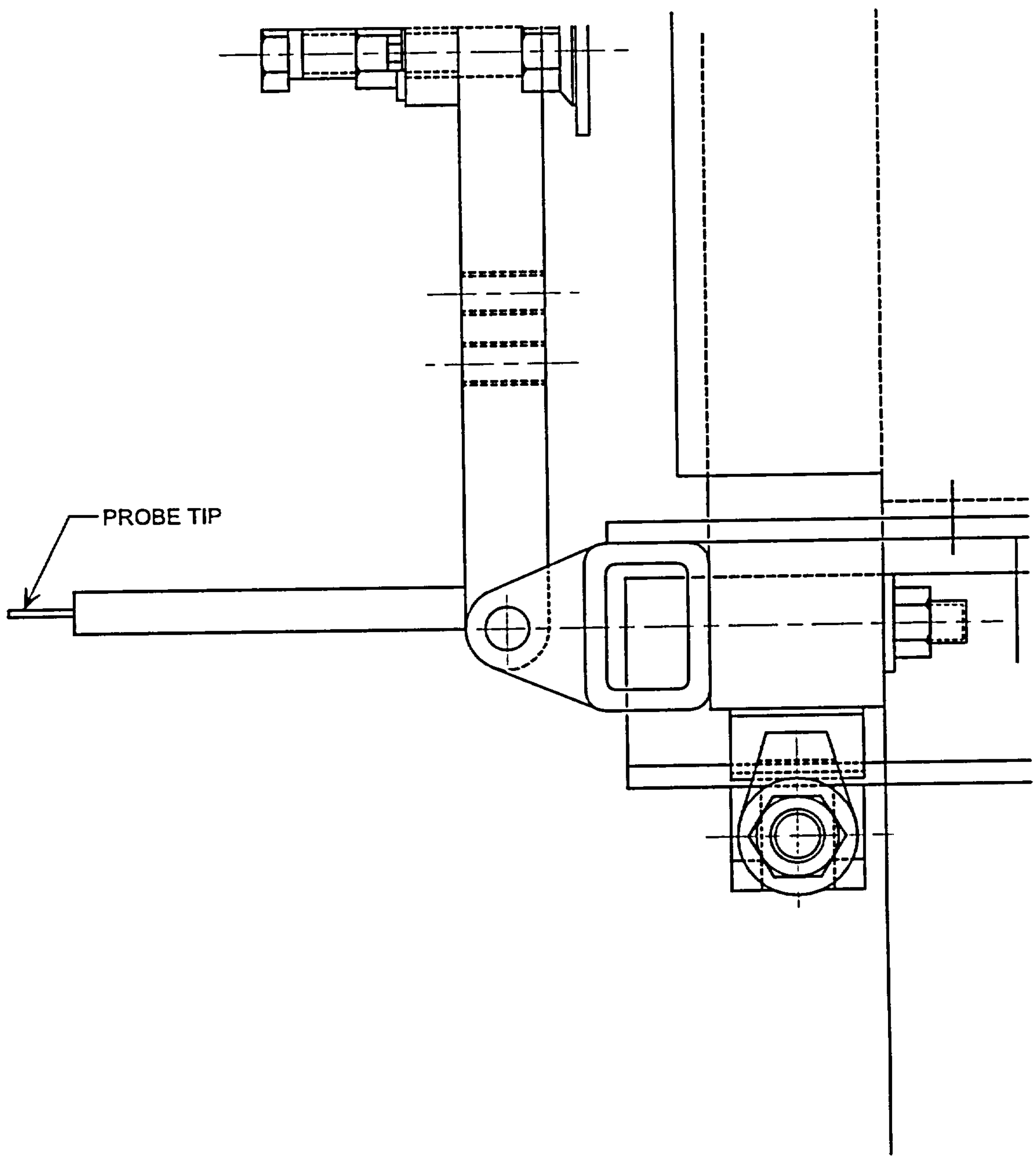

Figure 17.1-9 Overload Indicator

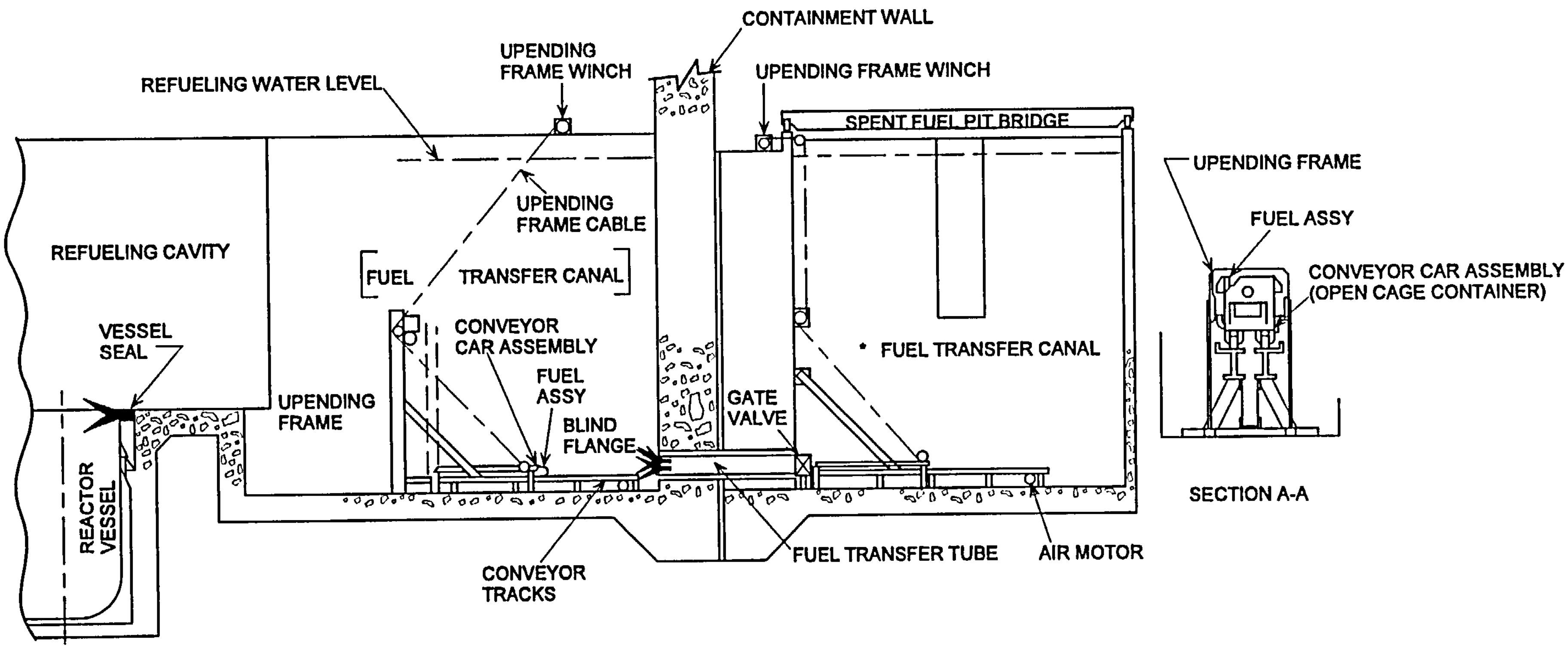

Figure 17.1-10 Fuel Transfer System

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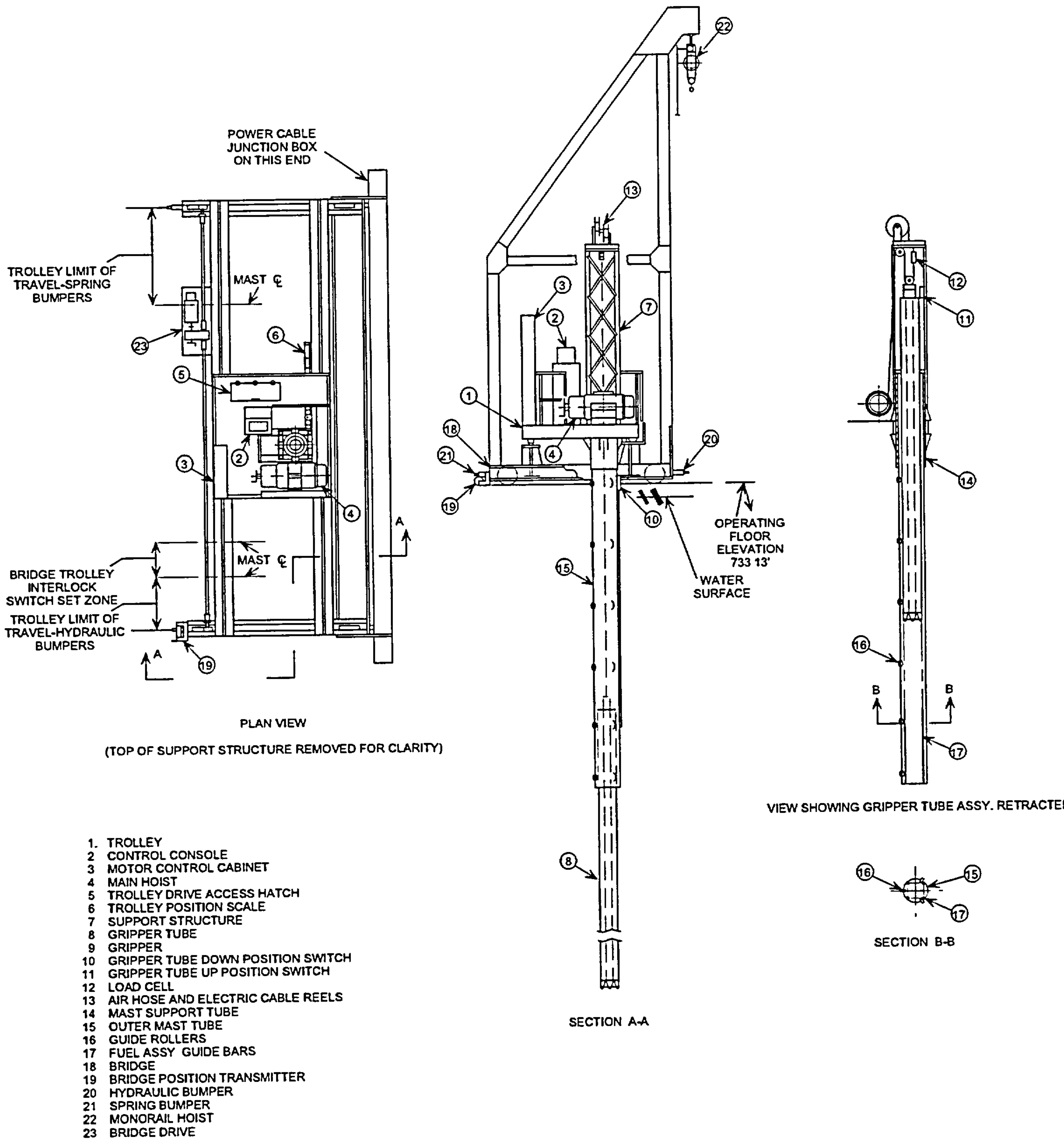

Figure 17.1-11 Manipulator Crane

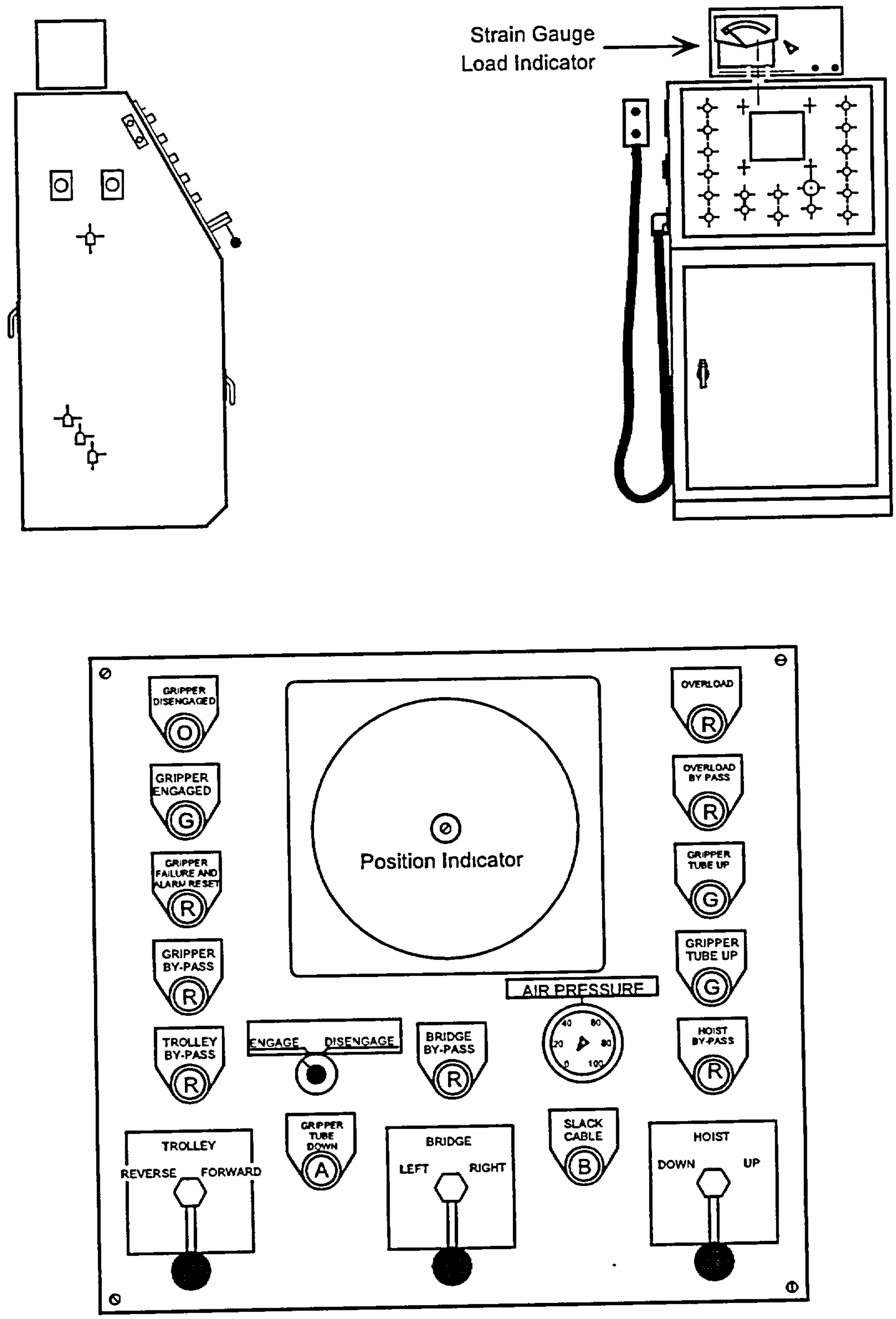

Figure 17.1-12 Manipulator Control Console

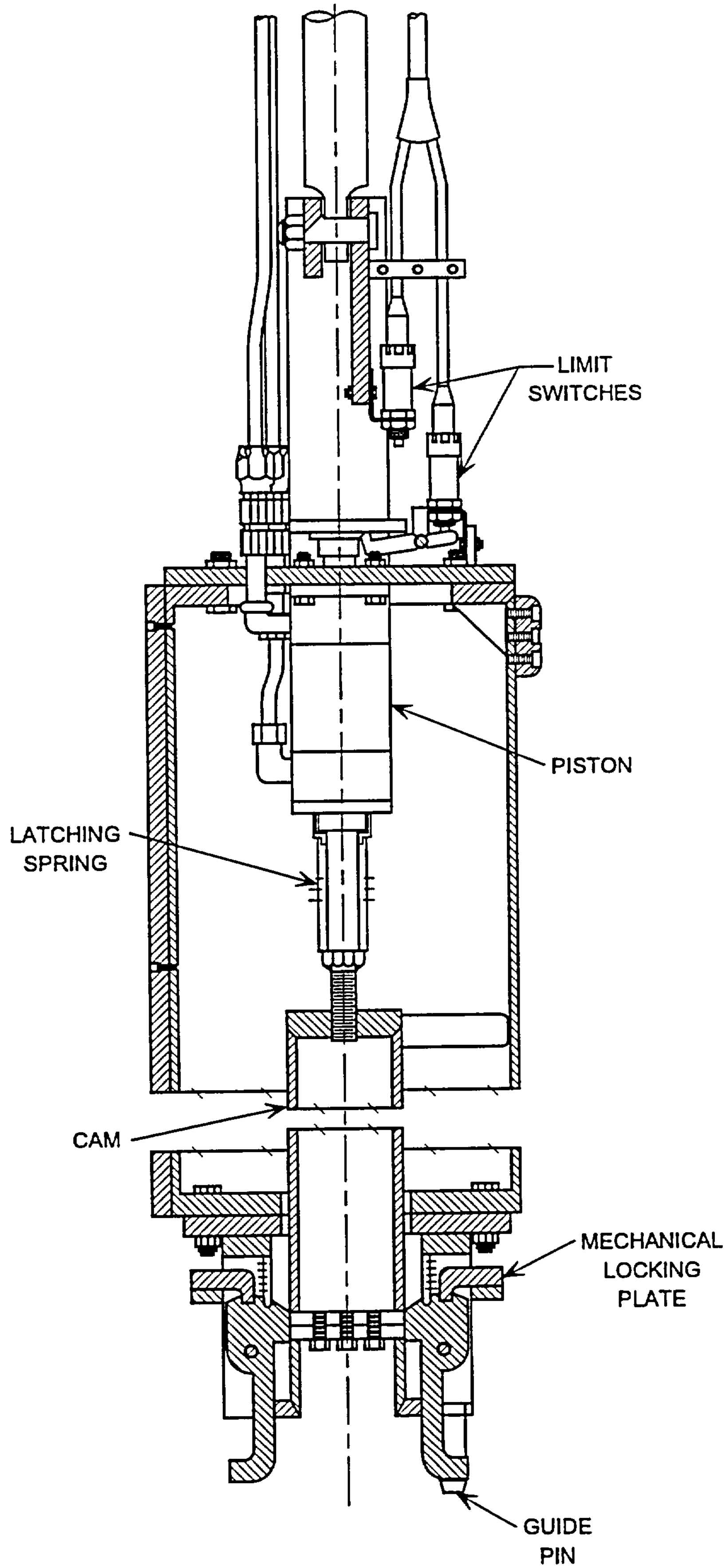

Figure 17.1-13 Gripper Assembly

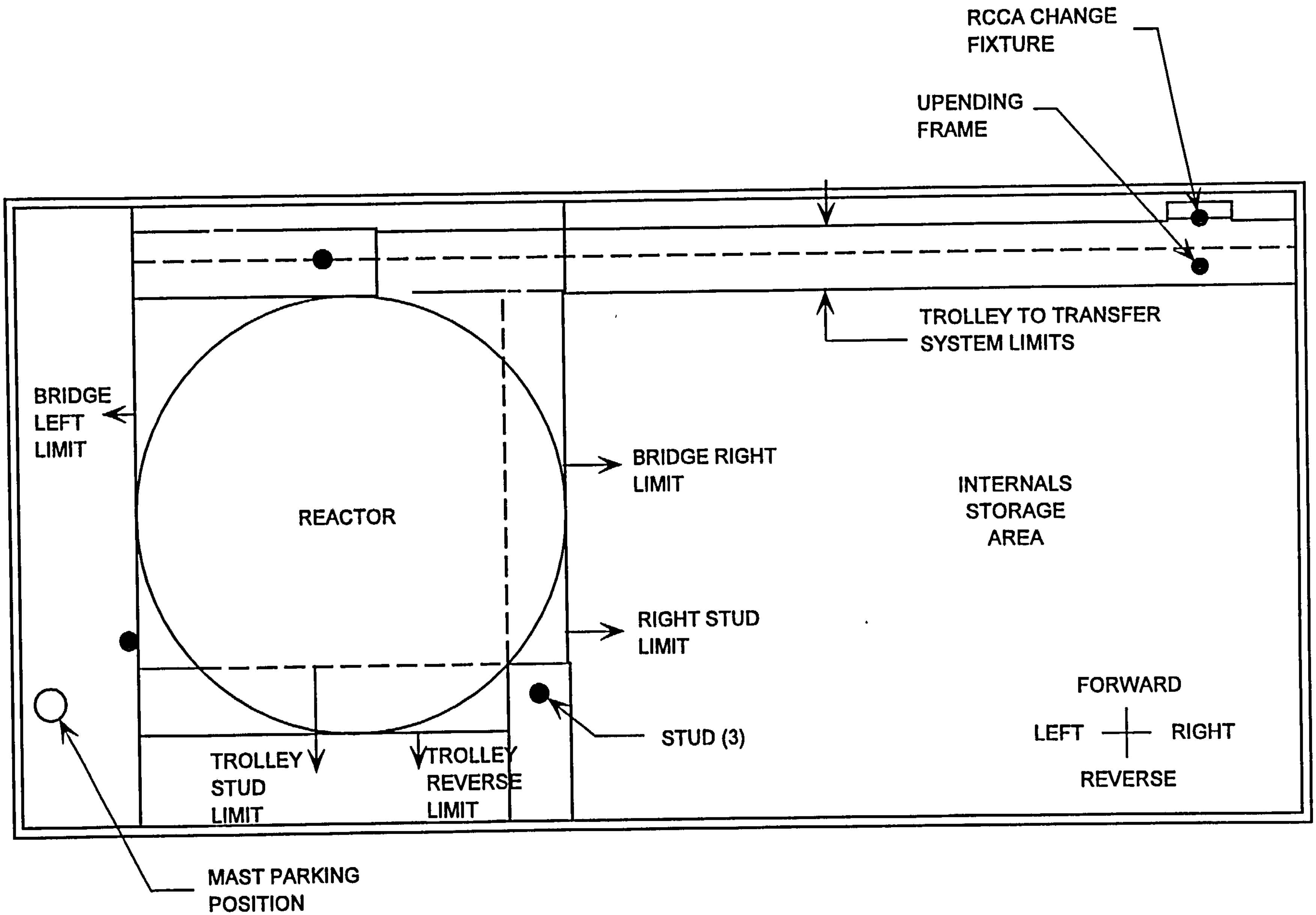

Figure 17.1-14 Manipulator Crane Travel Limits

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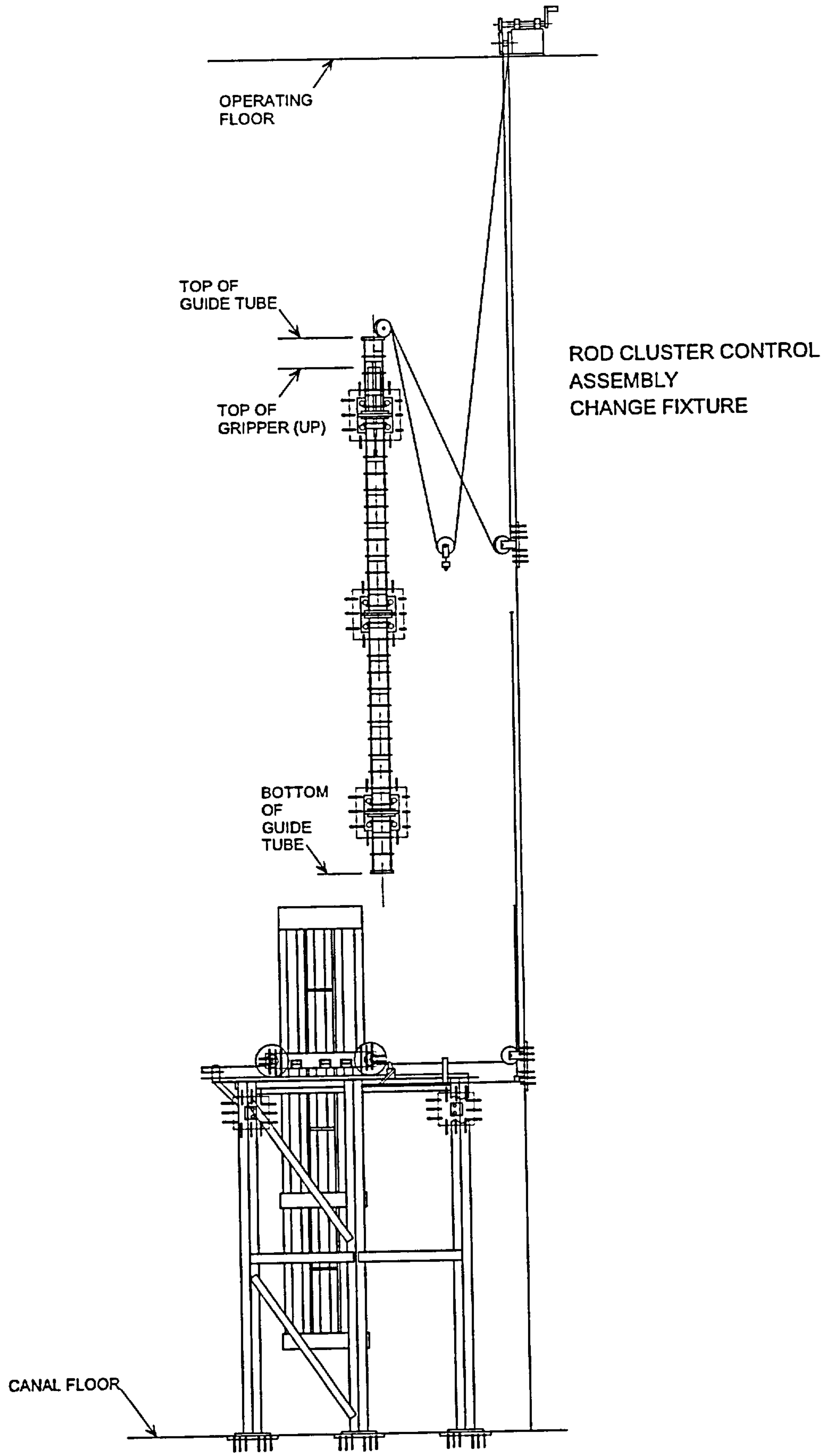

Figure 17.1-15 RCCA Change Fixture

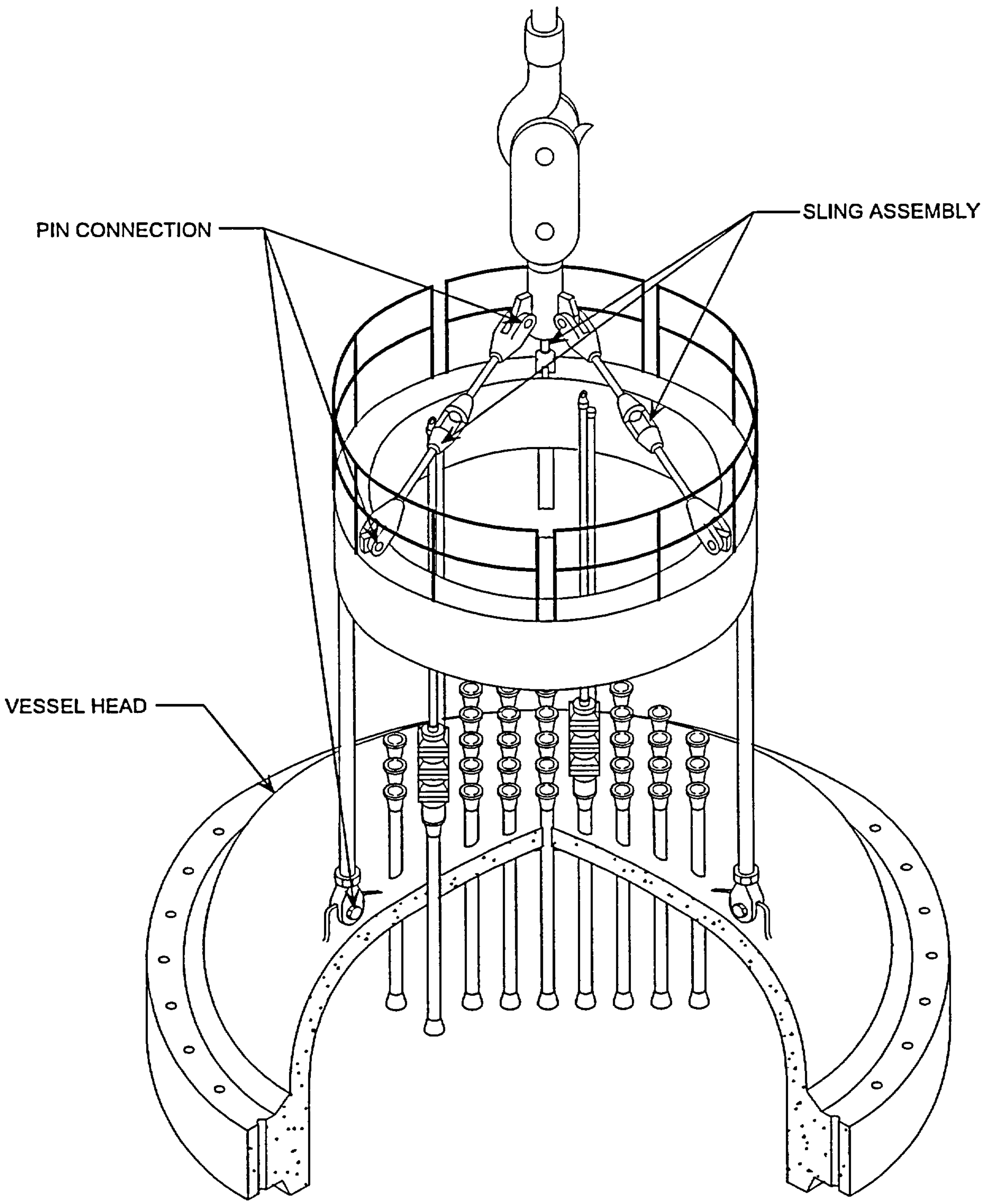

Figure 17.1-16 Reactor Vessel Head Lifting Device

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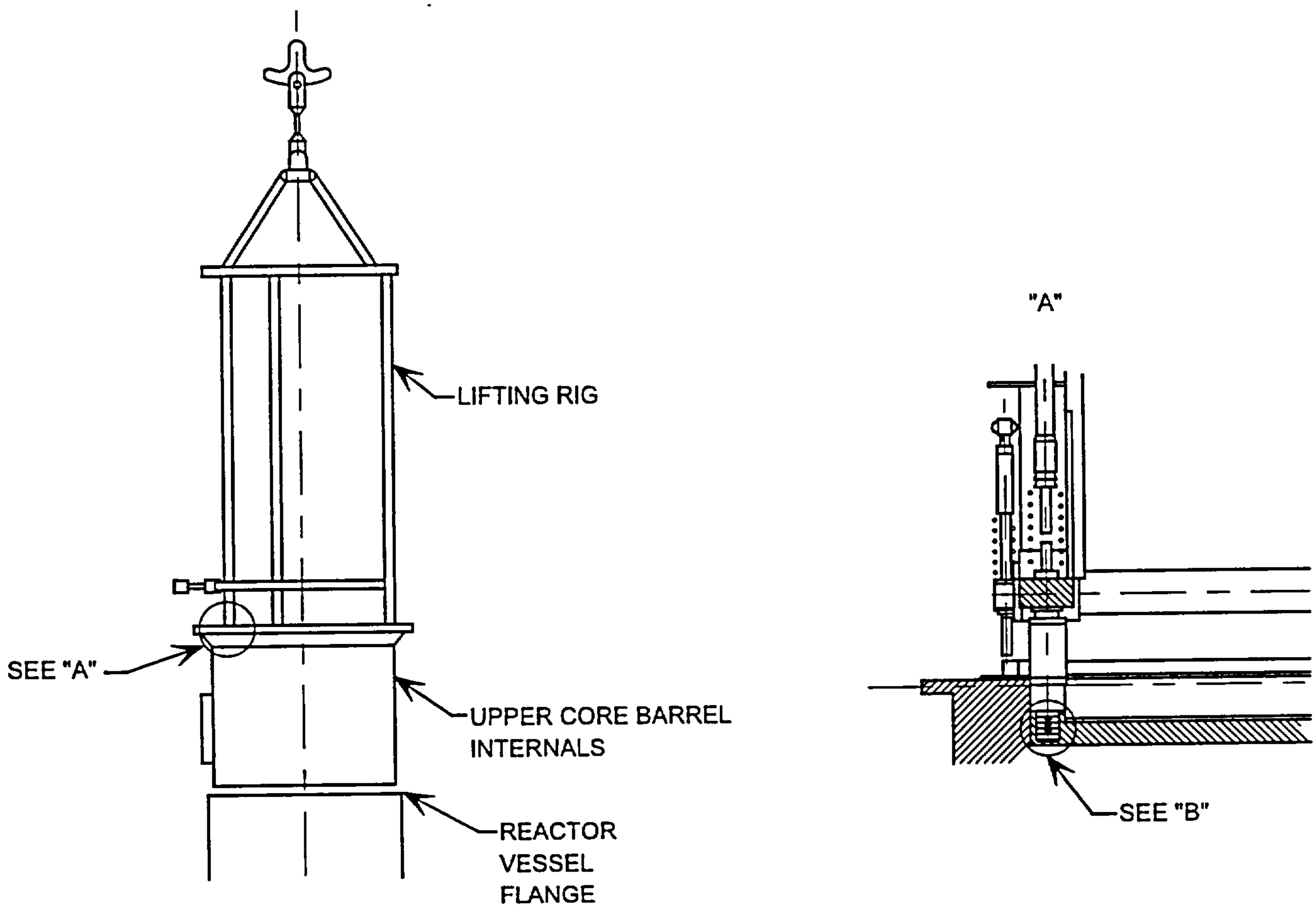

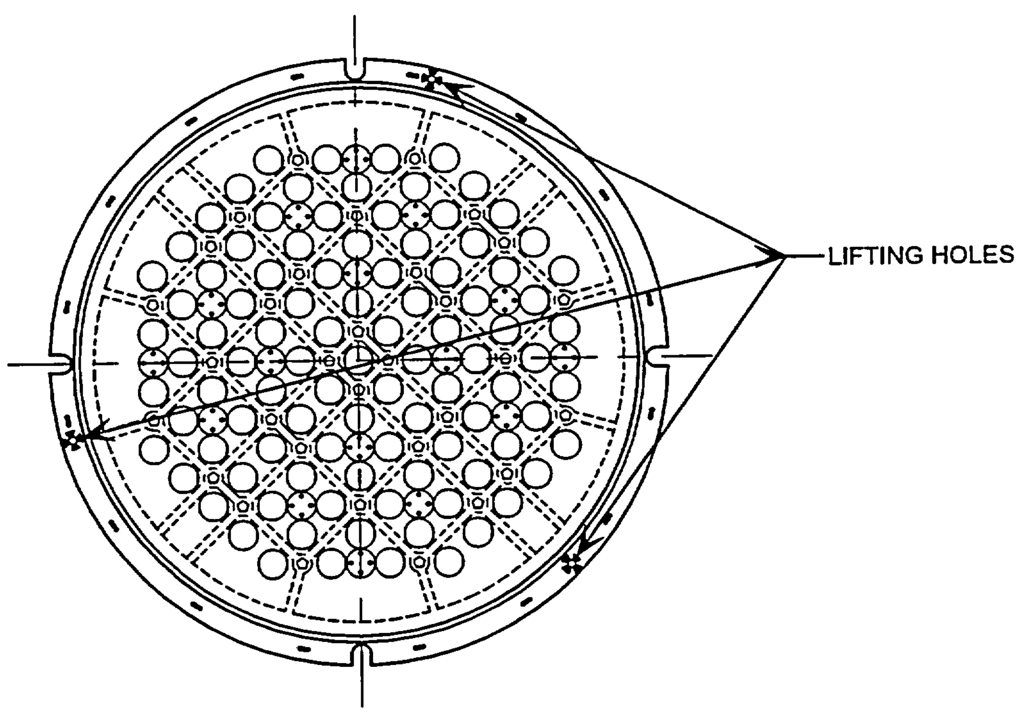

PLAN VIEW OF UPPER CORE SUPPORT STRUCTURE

Figure 17.1-17 Reactor Internals Lifting Device

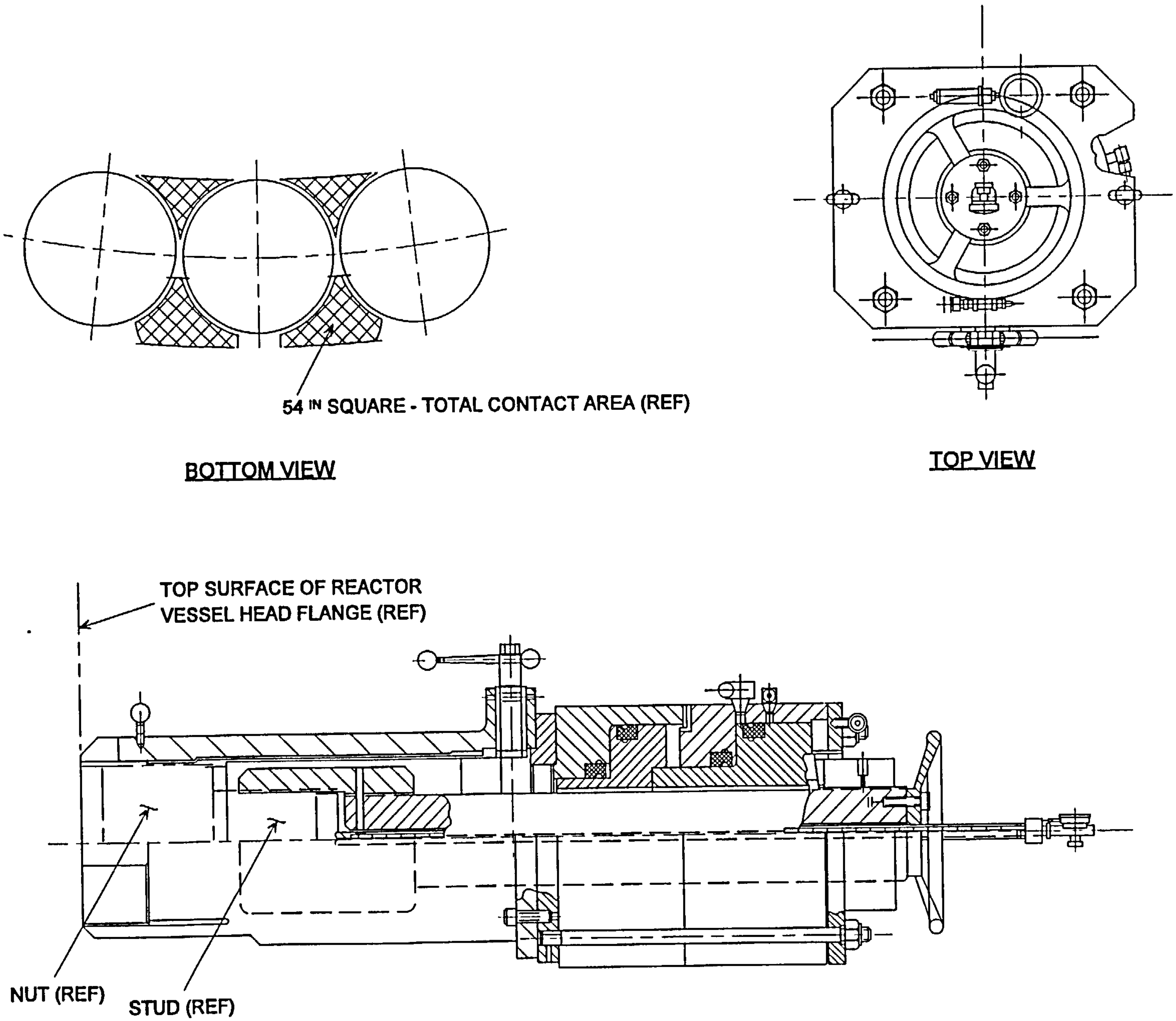

Figure 17.1-18 Reactor Vessel Stud Tensioner

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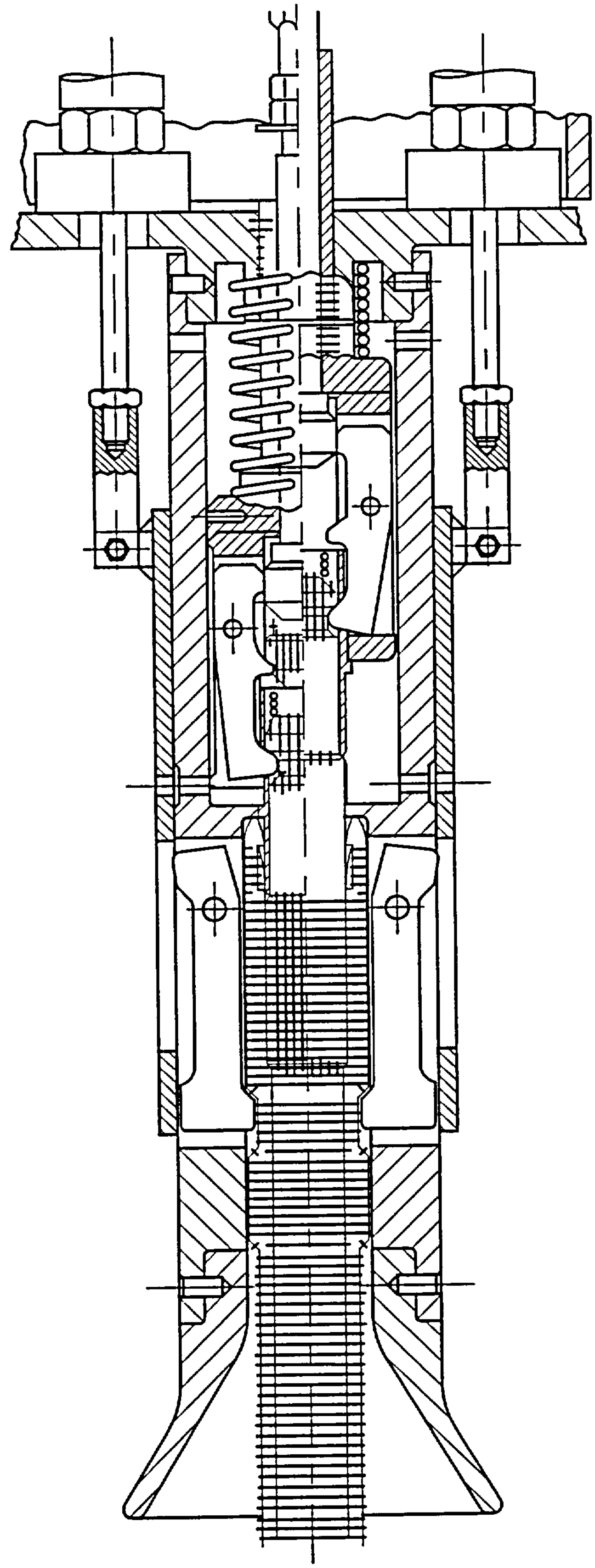

Figure 17.1-19 Rod Drive Shaft Unlatching Tool

Westinghouse Technology Systems Manual

Section 17.2

<u>Spent Fuel Storage</u>

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17.2 SPENT FUEL STORAGE

17.2.1 Introduction

Originally, the core was designed to be refueled with fuel assemblies containing recycled plutonium from spent fuel assemblies. Fuel assemblies would be sent to a facility where the plutonium and unused uranium would be chemically separated from the spent fuel and used in the manufacture of new assemblies. A demonstration plutonium extraction facility was designed and built, but fears associated with nuclear proliferation prevented the operation of the facility. The concern over the availability of relatively large quantities of weapons grade plutonium resulted in the U.S. government adopting policies which prohibited spent fuel assembly recycling. These policies forced the nuclear utilities to change spent fuel pool storage designs.

The original spent fuel pool storage design capacity was 4/3 core, i.e., if the core contained 200 fuel assemblies, the spent fuel pool storage could hold 266 spent fuel assemblies. This capacity would be sufficient for the 1/3 of the core's fuel assemblies that were changed out during a refueling plus a full core off load resulting from an unforeseen plant problem. Since spent fuel assemblies were to be recycled, this capacity was satisfactory. Also, the original spent fuel pool storage design had a 21 inch distance between the center line of adjacent storage rack locations. This distance was required to prevent spent fuel pool criticality under postulated worst case reactivity conditions. The 21 inch distance allowed the utilities to redesign the spent fuel racks. The large center line to center line distance was eliminated, and metal plates containing boron were added to prevent criticality. The second iteration of spent fuel pool storage racks was called high density fuel racks and more than doubled the spent fuel pool storage capacity. The change out of spent fuel pool storage rack's design is called "re-racking".

The increase in spent fuel pool capacity, combined with increases in core life, delayed the spent fuel storage problem. The arithmetic associated with spent fuel is not very complex. If 66 fuel assemblies are removed during each refueling, after 10 refuelings, the high density storage racks would be full. Many U.S. utilities have already re-racked twice. By the year 2000, it is estimated that there will be greater than 40,000 metric tons of spent fuel stored in the United States.

17.2.2 Dry Storage

Another option open to utilities to provide extra storage is dry storage. This allows spent fuel assemblies to be removed from the spent fuel pool and stored on-site in metal or concrete canisters on a concrete base.

Spent fuel assemblies are placed in the canisters while in the spent fuel pool and sealed. Because the oldest spent fuel would be placed in these canisters, the radioactivity and decay heat would be diminished, allowing the assemblies to be cooled by natural airflow around the outside of the canisters. Radiation from the canisters would be very low. At the boundary of the concrete base, there may be no detectable levels of radioactivity.

Dry storage entails the least amount of worker handling and therefore would result in less worker exposure to radioactivity. Canisters hold between 7 and 24 PWR assemblies each. Dry storage could open enough space in the spent fuel pool to accommodate plant needs into the twenty-first century. Since the NRC has licensed dry storage canisters (10CFR72), utilities would only need an additional license amendment from the agency. The NRC has licensed the designs of ten manufacturers; however, only four designs are currently in use. Table 17.2-1 lists the licensed manufacturers and table 17.2-2 lists the utilities that use dry storage. The major features of the one of the four designs will be discussed in the following paragraphs.

17.2.2.1 NUHOMS

The Pacific Nuclear Fuel Services, Incorporated dry storage system is called NUHOMS and is of the concrete modular design. As shown in Figure 17.2-1, the design consists of a dry shielded canister (DSC) and a reinforced concrete horizontal storage module. The DSC serves as the containment pressure boundary for the confinement of radioactive materials and provides a leak tight, inert atmosphere to ensure that the integrity of the fuel cladding is maintained.

The DSC is composed of two basic components: the basket and the shell. The DSC basket is designed to accommodate 24 fuel assemblies (Figure 17.2-2) in separate rectangular boxes. The 24 boxes are placed into spacer disks separated by support rods. The spacer disks allow the rectangular boxes to be placed into the circular shell.

The DSC shell assembly is a stainless steel, welded pressure vessel that provides a leak tight confinement barrier for all radioactive material and envelopes the spent fuel assemblies in an inert helium atmosphere. The DSC cylindrical shell and cover plates encapsulate the basket assembly holding the spent fuel. The DSC is placed in a transfer cask for loading, and it will remain in the cask until it is transferred into the horizontal storage module (HSM).

The ventilated reinforced concrete HSM is the principal structure that is located on each Independent Spent Fuel Storage Installation (ISFSI) site. Each HSM (Figure 17.2-3) is a free-standing prefabricated unit placed on a non-safety related basemat. Each HSM is designed to provide storage for one DSC. Ports in the bottom of the HSM allow for the passage of air into the vicinity of the DSC for cooling purposes. Heated air passes from the inside of the HSM to the environment via ports in the top of the HSM. HSMs are placed side by side until the desired number of fuel assemblies are stored.

The basic steps in the transfer of spent fuel from the spent fuel pool to the HSM are:

1. Clean and load the DSC into the transfer cask.
2. Fill the DSC and cask with water and install the cask/DSC annulus seal.
3. Place the transfer cask containing the DSC in the spent fuel pool.
4. Load the spent fuel assemblies into the DSC.
5. Place the top shield plug on the DSC.
6. Remove the loaded cask from the spent fuel pool and place it in the decontamination area.
7. Lower the water level in the DSC below the shield plug.
8. Place and weld the inner top cover to the DSC shell and perform NDE.
9. Drain the water from the cask/DSC annulus.
10. Drain the water from the DSC.
11. Evacuate and dry the DSC.
12. Fill the DSC with helium.
13. Perform a helium leak test on the seal weld.
14. Seal weld the siphon and vent port plugs and perform NDE.
15. Fit -up the outer top cover plate with the DSC shell.
16. Weld the top cover plate to the DSC shell and perform NDE.
17. Install the transfer cask top cover plate.
18. Lift and downend the transfer cask onto the transport trailer.
19. Ready the HSM to receive the DSC.

20. Ready the cask for transport and to the transport trailer to the HSM.
21. Position the transfer cask with the HSM access opening.
22. Remove the transfer cask top cover plate.
23. Align and secure the transfer cask to the HSM.
24. Set up for the DSC transfer.
25. Place the DSC into the HSM.
26. Disengage the transfer cask from the HSM.
27. Install the HSM door.

17.2.3 Pin Consolidation

The capacity of the spent fuel pool could also be expanded by pin consolidation, in which the fuel rods (or pins) within each fuel assembly are reconfigured to reduce the amount of space required for storage. The fuel pins in the assemblies are removed and placed in new "consolidation" cages which allow the rods to be stored closer together. The original end caps and grids are sheared to reduce their volume and are also stored in the spent fuel pool.

Several U.S. nuclear plants have performed pin consolidation demonstration projects while completing their own study of options for storage expansion. Because pin consolidation is very labor intensive, worker exposure to radioactivity must be taken into account when considering this storage option. There are ways to reduce exposure, such as consolidating older fuel first, since this fuel is less radioactive. With pin consolidation, all spent fuel generated during the remainder of the plant's licensed operating life could be safely stored within the spent fuel pool. Pin consolidation would require license amendments from the NRC.

17.2.4 Summary

The initial design of the spent fuel pool provided limited storage capacity. Political decisions dealing with fuel recycling forced the utilities to face the dilemma of storing fuel on site. Storage capacity has been increased by two methods: re-racking and dry storage. Re-racking involves using fuel storage racks with reduced distances between adjacent fuel assemblies. Dry storage involves transferring spent fuel to a shielded cask and placing the shielded cask in a concrete vault located on the plant site. The feasibility of pin consolidation has been demonstrated but is not currently being used.

Table 17.2-1 NRC Approved Dry Spent Fuel Storage Designs

Vendor	Design Model	Capacity
General Nuclear Systems, Inc.	Metal Cask CASTOR V/21	21 PWR Fuel Assemblies
Pacific Nuclear Fuel Services, Inc.	Concrete Module NUHOMS-7	7 PWR Fuel Assemblies
Pacific Nuclear Fuel Services, Inc.	Concrete Module NUHOMS-24P	24 PWR Fuel Assemblies
Foster Wheeler Energy Applications, Inc.	Concrete Vault, Module Vault Dry Stone	83 PWR or 150 BWR Fuel Assemblies
Nuclear Assurance Corporation	Metal Cask NAC-STC	26 PWR Fuel Assemblies
Nuclear Assurance Corporation	Metal Cask NAC-128/ST	28 PWR Fuel Assemblies
Nuclear Assurance Corporation	Metal Cask NAC-C28T	28 Canisters (fuel rods from 56 PWR assemblies)
Transnuclear Incorporated	Metal Cask TN-24	24 PWR Fuel Assemblies
Pacific Sierra	Concrete Cask VSC-24	24 PWR Fuel Assemblies
Westinghouse Electric	Metal Cask MC-10	24 PWR Fuel Assemblies

Table 17.2-2 NRC Approved Dry Spent Fuel Storage Designs

Plant	NSSS Vendor	Storage Vendor	Storage Model
Surry 1,2	Westinghouse	General Nuclear Systems, Inc.	CASTOR V/21
H.B. Robinson	Westinghouse	Pacific Nuclear Fuel Services, Inc.	NUHOMS-7
Oconee 1,2,3	Babcock and Wilcox	Pacific Nuclear Fuel Services, Inc.	NUHOMS24P
Fort St. Vrain	General Atomics	Foster Wheeler Energy Applications, Inc.	Module Vault Dry Stone
Calvert Cliffs 1,2	Combustion Engineering	Pacific Nuclear Fuel Services, Inc.	NUHOMS-24P
Brunswick 1,2	General Electric	Pacific Nuclear Fuel Services, Inc.	NUHOMS-7
Prairie Island	Westinghouse	Transnuclear Incorporated	TN

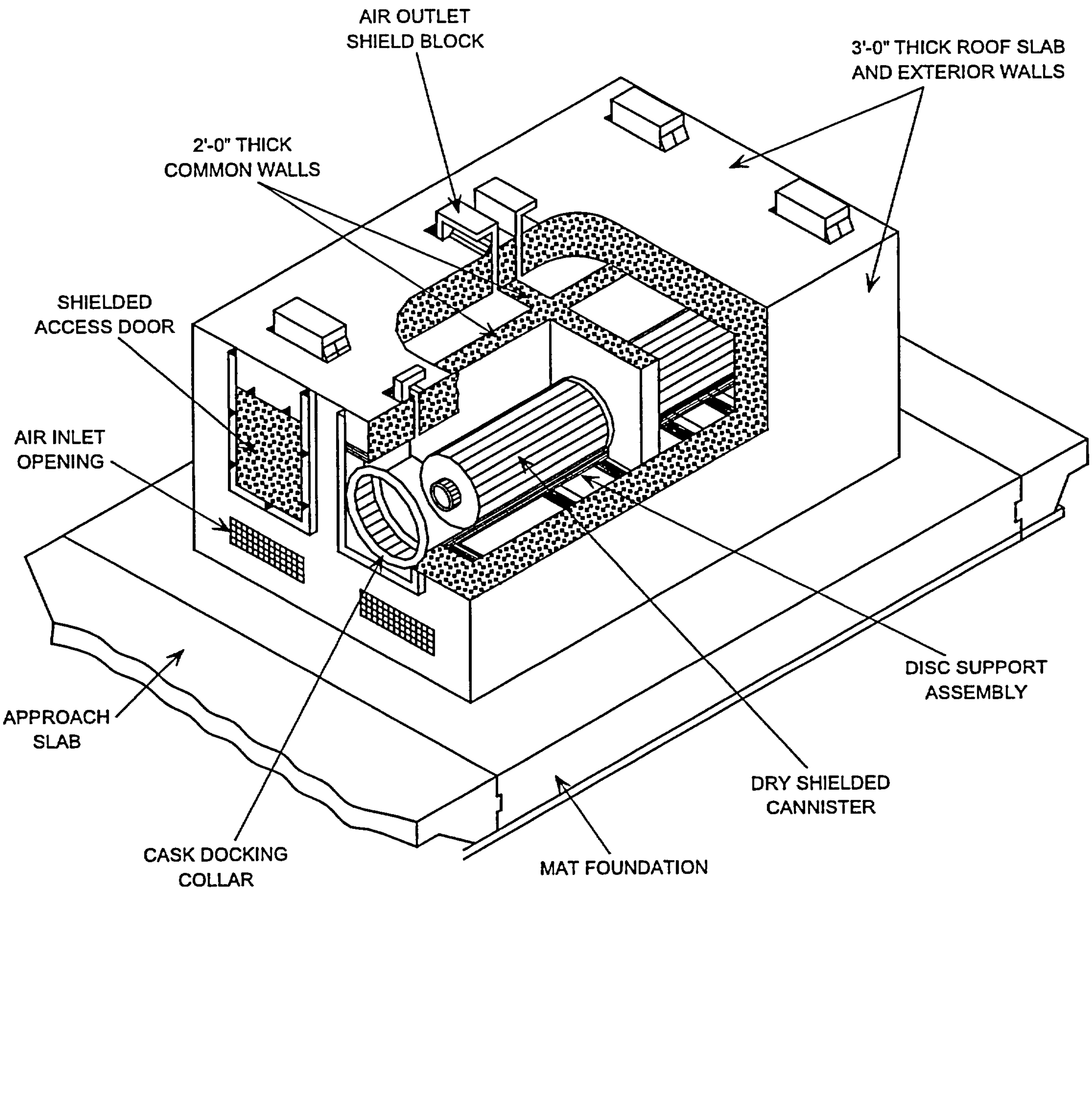

Figure 17.2-1 NUHOMS

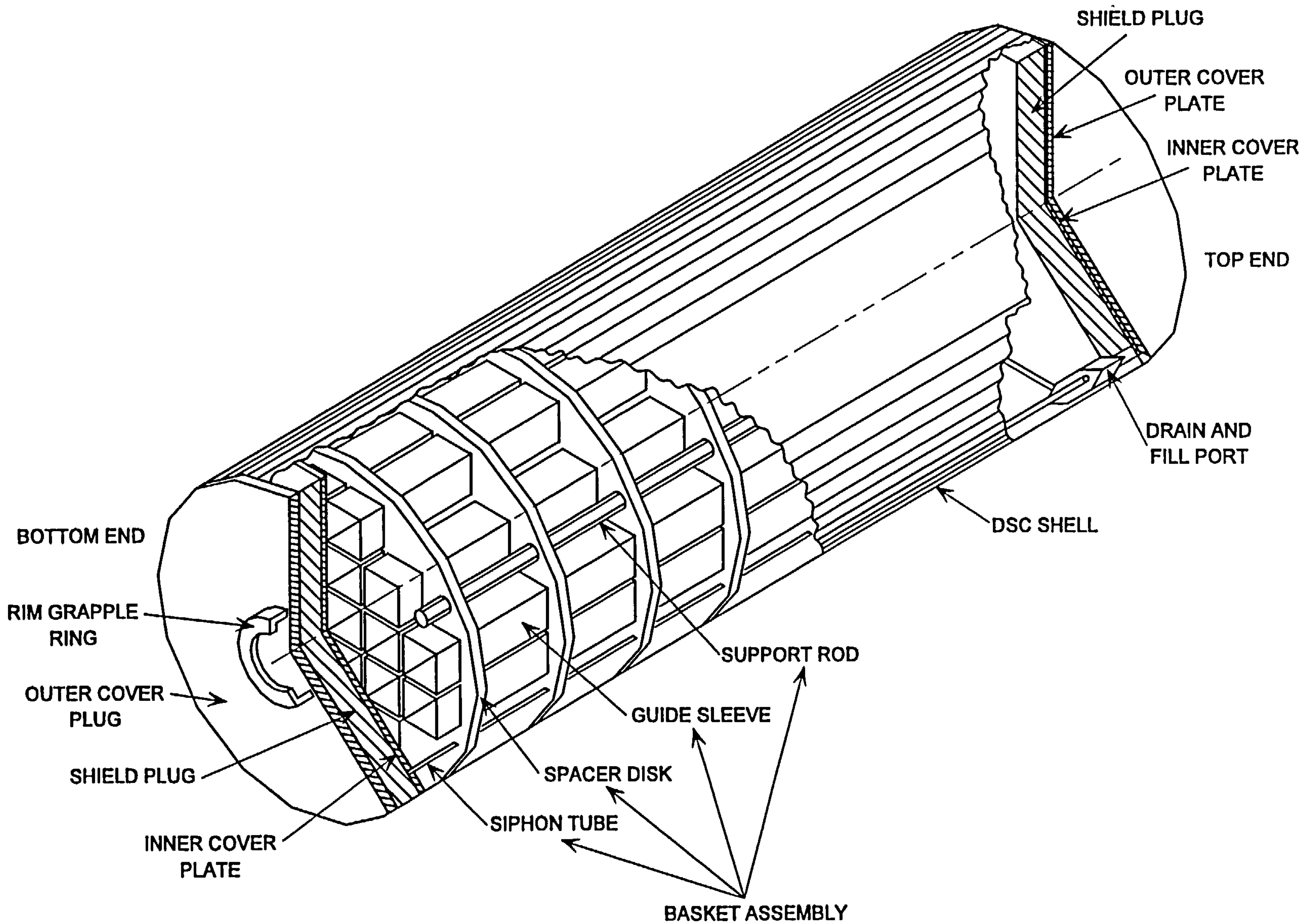

Figure 17.2-2 Dry Shielded Cannister

0199-3

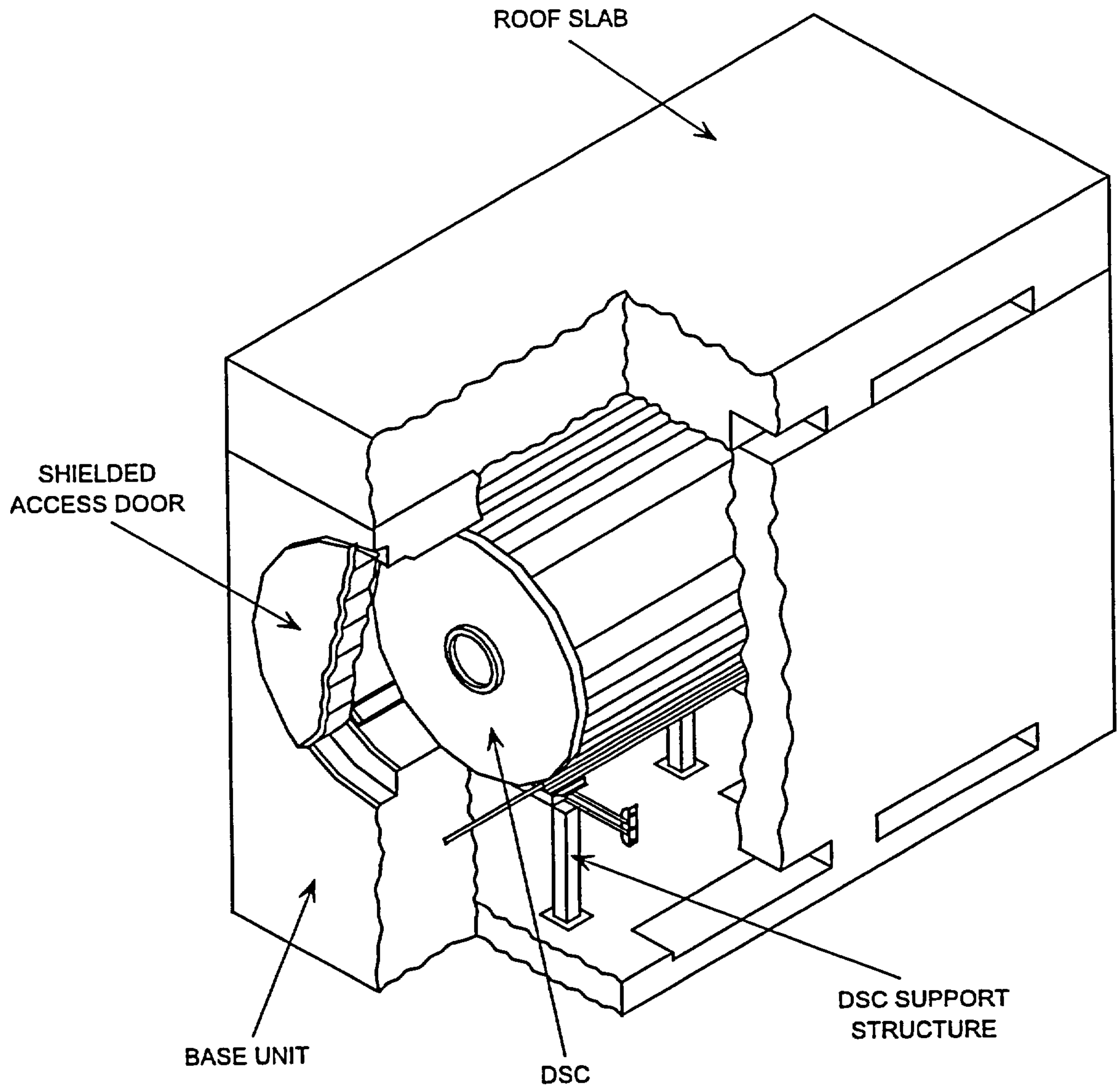

Figure 17.2-3 Horizontal Shielded Module

Westinghouse Technology Systems Manual

Chapter 18

PLANT COMPUTER

Section

Westinghouse Technology Systems Manual

Section 18.0

Plant Computer

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18.0 PLANT COMPUTER

Learning Objectives:

After studying this chapter you should be able to:

1. State the purpose of the plant computer.

2. Briefly describe the purpose of the following programs and the types of information available from each.

   a. Sequence of events
   b. Post trip review
   c. Periodic and demand log
   d. Incore data collection and reduction
   e. NSSS process supervision
   f. Reactor Control and Protection System supervision

18.0.1 Introduction

The purpose of the plant computer is to provide on-line data acquisition, alarming, logging, and data reduction to aid the operator in optimizing plant performance. Although the computer can be a valuable operational tool, the plant is designed to be operable when the computer is not available.

The base computer is for the primary side of the nuclear plant, but some secondary plant inputs are needed for primary plant programs. There are optional computer capabilities which can be incorporated into the system to monitor additional secondary plant parameters.

18.0.2 System Description

The generation of plant computers used in most operating Westinghouse plants is the PRODAC 250, a high speed digital computer. The equipment of concern to the operator consists of three or more output typewriters, six 2-pen trend recorders, and the operator's control panel with two 6-character visual displays and pushbuttons. All of this equipment is located in the main control room. The central processor unit, random access disc memory, programmer's console, and other necessary equipment are located in the computer room. Figure 18.0-1 shows the control room equipment and a close-up of the control panel.

The functions of the computer are:

1. To provide for continuous monitoring of plant variables and alarm of off-normal conditions.

2. To provide a record of related variables when certain off-normal conditions occur, including reactor trips.

3. To provide on-line data reduction for certain parameters to assist the operator.

4. To provide data to be used in off-line data reduction in larger offsite computers.

The computer periodically scans analog and contact inputs. Some contact inputs are monitored continuously and are classified as interrupts. All inputs are stored in the computer to form a data base for other computer functions. The following summarizes more precisely the functions performed by the computer systems:

General Analog and Contact Input Supervision

- Analog scanning and conversion
- Contact input scanning
- Alarming
- Analog trend
- Digital trend
- Visual Display
- Post trip review
- Sequence of events recording
- Sensor calibration
- Periodic and demand log
- Summary log
- Operator-computer communications

Reactor Control and Protection System Supervision

- Reactor coolant average loop temperature setpoint
- Pressurizer level controller setpoint
- Control rod cluster position deviation
- Reactor protection system
- Deviation in redundant measurements

Nuclear Steam Supply System Process Supervision

- Overpower and overtemperature trip setpoint
- Calibration check of power range channel signals
- Steam generator total thermal output
- Reactor thermal output
- Reactor dynamic thermal output
- Unit net efficiency
- Tilting factors
- Load follow
- On-site, incore data collection
- On-site, incore data reduction

The important and outstanding features of the application-oriented programs to perform the above functions are described in the following paragraphs.

18.0.2.1 General Analog and Contact Input Supervision

18.0.2.1.1 Analog Scanning and Conversion

In general, the analog scan program commands the computer and the analog input subsystems to: read the value of all analog inputs in a pre-established manner; check the reasonableness of readings; convert the values into engineering units and store them at predetermined core locations; maintain or update information words which describe the condition of each input (i.e., in alarm condition, removed from scan, out of range, etc.); and initiate various programs.

18.0.2.1.2 Contact Input Scanning

All contact inputs are scanned at least once every two seconds, and their status is stored in computer memory for used by various other programs.

Contacts requiring immediate attention (for example, those included in the sequence of events program) are classified as "interrupt" inputs. Such interrupt inputs are not scanned periodically; rather, their changes of state receive immediate attention. Provisions are included to initiate programs for selected contact and interrupt inputs on bidirectional changes of input state.

18.0.2.1.3 Alarming

The analog and contact input alarm program compares the values of analog inputs and the status of contact inputs against constant or variable alarm limits, and warns the operator when off-normal conditions exist by actuating an audible alarm and printing out an alarm message (red) on the operational typewriter. A return to normal message (black) is printed out when the input value returns below its alarm limit.

18.0.2.1.4 Analog Trend

Up to twelve addressable points may be selected for simultaneous analog trending. The operator's (Figure 18.0-1) panel provides for selecting the desired inputs, starting and stopping the trending, and selecting a suitable span and offset. This analog trend functions can be used to record suspected fluctuations in any measurement or to obtain data for future analysis of transients during plant startup or load changing. Inputs selected for the analog trend will be put on four-second scan by the program and the analog output will be updated every four seconds. In addition, the operator can receive an operational typewriter printout of analog trend information pertinent to the inputs being trended.

18.0.2.1.5 Digital Trend

It is possible to trend any analog input or addressable calculated value on the trend typewriter. Up to six groups of 20 variables may be assembled through the operator's control panel. Any one group may be selected for trending. Variables to be trended and trending intervals are selectable through the operator's console. All points are trendable at the last selected interval. (Intervals are 1 to 99 min.; however, the scan frequency of the selected inputs is not changes.)

18.0.2.1.6 Visual Display

Two sets of six-digit visual displays are provided on the operator's panel (Figure 18.0-1). Any analog input or addressable value may be displayed in engineering units on this display. One such unit displays the input identification (address), the other displays the value. The program updates the display windows every four seconds.

18.0.2.1.7 Post Trip Reviews

Post trip review programs retain the values of two groups of selected analog inputs prior to and after trip, or whenever demanded by the operator. The printed-out values may be used for analysis of plant behavior.

One group of 84 selected inputs will be recorded every eight seconds during the two minutes prior to trip, and then every eight seconds during the first three minutes following the trip. The second group, consisting of up to 12 selected inputs, will be recorded every two seconds during the eight seconds prior to trip, and every two seconds during the first eight seconds following the trip.

If a plant interrupt occurs while an operator-initiated printout is in progress, the printout will be terminated and the actual post trip printout will begin.

The pre and post trip information is printed out on the trend typewriter. A current trend is interrupted for the duration of the pre/post trip data, and the trend will be automatically restored after completion of the pre/post trip data printout. An example of a post trip review is shown in Figure 18.0-2a & b.

The beginning of a post trip review is shown in Figure 18.0-2 (a and b). The times at the left are actual time of the printout. At the head of each column is the address of the parameter, and the list of values down to the next address is a sequential list of the parameter at certain intervals. The time for each value is in the first time column to the right. The time is in minutes, seconds, and tenths of a second. The time of trip is listed in hours, minutes, and seconds. Parameters are grouped and use a common time column.

18.0.2.1.8 Sequence of Events Recording

This program records the sequence of status change of up to 40 plant contacts either directly or closely associated with tripping the reactor. Knowing the occurrence of these events will contribute to analyzing the causes and effects of reactor trips and assist in trouble-shooting.

Both the contact closures and contact openings are required as status changes. The time between the occurrence of the first and the subsequent status changes is determined to the nearest cycle (16.6 milliseconds at 60 cycles). The order of occurrence of events which are four or more milliseconds apart are determined by the program. Even if more than one status change occurs within less than four milliseconds, the events are still detected and recorded.

Figure 18.0-3 shows several sequence of events records. The time at the left is the time of the printout. The time of the first event is given in hours, minutes, and seconds. The status changes are then given sequentially beginning with the first event at cycle (C) o. The address and name of each parameters is given with its status and time since first event in cycles. Sixty cycles equals 1 second.

18.0.2.1.9 Sensor Calibration

When the computer reads an analog input and converts the millivolt or volt reading into engineering units, it acts as a measuring instrument. The conversion of the value to engineering units is based on a polynomial equation representing the millivolt/ volts vs. engineering value. The constants of this polynomial are stored in the computer memory, one set for each input.

If the output signal of the sensor changes with respect to the volt vs. engineering value, either as a result of maintenance action or deterioration, the "conversion curve" stored in the computer memory must also be changed. Plant maintenance personnel can thus replace the old calibration information with the new, corrected information through the operator's console pushbutton controls.

A Least Squares Program may also be initiated through the operator's console. The program computes a polynomial curve for a set of data points, using the "least squares" technique. The new calibration curve thus generated has a minimum mean-square error between the data points and the corresponding points on the curve. The program can be employed to determine a new set of conversion constants for one of the analog inputs in the system, or to fit general data to obtain a general curve fit (e.g., a bogey curve).

18.0.2.1.10 Periodic and Demand Log

The values of up to 150 analog inputs and computed results may be recorded either periodically or on demand on preprinted "roller- towel" type forms. The periodic log interval is selectable (30 to 60 min.) through the operator's console. In addition, the operator may demand a printout between the above periodic logs. Figure 18.0-4 shows part of a periodic log.

18.0.2.1.11 Summary Log

A summary of the daily log entries is automatically initiated on the logging typewriter at the end of each 24 hours.

18.0.2.1.12 Operator-Computer Communications

The operator's console provides for executing the following functions by pushbutton controls:

1. Alarm Limits
   - change upper alarm limit (key-locked)
   - change lower alarm limit (key-locked)
   - add/omit input from limit checking
   - print alarm limits
2. Alarm review, including:
   - analog and contact inputs which are in alarm condition
   - review of setpoint supervision alarms
   - review of rod cluster deviation alarms
   - review of deviating inputs from redundant measurements
3. Digital trending on typewriter:
   - add and delete points
   - select trending intervals
   - select one of six groups
4. Analog trending
   - select span
   - select offset
   - select pen
5. Visual display
6. Demand post trip review
7. Demand periodic log
8. Add a point to scan (key-locked)
9. Remove a point from scan (key-locked)

10. Enter value or constant required for calcula tions (key-locked)
11. Demand instantaneous printout of a single analog input value
12. Demand last stored value printout for all addressable inputs:
    - analog input
    - specified calculated values
    - constants
    - addressable average and accumulated values
    - digital input status
13. Demand certain performance calculations
14. Update digital contact pulse input counters
15. Summary reviews
    - points removed from scan by the operator
    - points out of range
    - print all analog values
    - print all digital input states
    - print all alarm limits

Demand printout of all analog inputs being trended on the analog trend recorder, including all pertinent trending information.

18.0.2.2 Reactor Control and Protection System Supervision

18.0.2.2.1 Reactor Coolant Average Loop Temperature Setpoint Supervision

This program computes the expected setpoint of the reactor coolant average loop temperature controller, compares it with the actual setpoint, and then warns the operator if the deviation exceeds preset limits. The expected value of the setpoint is calculated as a function of load (or turbine first stage pressure). This relation is stored in the computer memory in the form of a polynomial.

18.0.2.2.2 Pressurizer Level Controller Setpoint Supervision

This program computes the expected setpoint of the pressurizer level controller, compares it with the actual setpoint, and then warns the operator if the deviation exceeds preset limits. The expected value of the setpoint is calculated as a linear function of the average loop temperatures. This relation is stored in the computer memory in the form of a polynomial.

18.0.2.2.3 Control Rod Cluster Position Deviation Alarms

This program periodically checks the control rod cluster positions by comparing rod cluster analog position measurements with the position of the respective rod bank drive. Since all rod clusters are moved in groups by bank, individual rod cluster analog measurements should agree with the bank drive movement. Digital inputs representing one step up or down of the bank driving mechanism are counted and the net total is compared with the analog position of the rod cluster in the bank. Deviation of rod cluster positions by more than a preset amount from the corresponding bank rod measurement initiates an alarm. Deviation among rod cluster positions within the common group and deviation among group positions with the common bank by more than a preset amount also initiates an alarm. Some examples of alarm messages are shown in Figure 18.0-5 (a and b).

18.0.2.2.4 Digital Trend and Monitoring of the Reactor Protection Systems

The purpose of this program is to show the relative rates of change of the most important measurements associated with partial tripping of the reactor, and the maximum value that each such variable reaches after an alarm. The program will also show the status of the partial trip channel bistables and the cause (output) of trip coincidence logic with the corresponding channel variable.

When any one of the digital input interrupts from the digital trip channel bistables exceeds its trip setpoint, and/or any one of the digital inputs from the cause of trip coincidence logic actuates, and/or any one of the participating analog inputs exceeds its alarm limits, this program is automatically initiated.

When initiated, the program will collect instantaneous values of pertinent analog inputs and will print out on the Operational typewriter the address and value of these inputs. The typewriter will also print out the address and state of partial trip digital inputs. An example is shown in Figure 18.0-6.

18.0.2.2.5 Deviation in Redundant Measurements

This program is designed to assist the plant operator and contribute to the safety of the plant. It may also be arranged to provide a history of the operating reliability and accuracy of the redundant measurements. For each redundant measurement, the computer compares the analog inputs, each against the others. If the deviation exceeds a preset value, an appropriate alarm is initiated. A "Q" is printed next to a value which is outside a preset cluster limit. An "*" indicates an unreliable value. Measurements that are thus processed come mainly from the reactor coolant systems.

18.0.2.3 Nuclear Steam Supply System Process Supervision

18.0.2.3.1 Overpower and Overtemperature Trip Setpoint

These setpoints are calculated by the computer and compared with the respective trip setpoints calculated by conventional analog instrumentation. The computer will alarm if the setpoint deviation exceeds preset limits.

18.0.2.3.2 Calibration Check of Power Range Channel Signals

This program will warn the plant operator if nuclear power range channel readings deviate during steady-state operation from the calculated Reactor Dynamic Thermal Output (RDTO) and the Reactor Thermal Output (RTO), or during transient operation from the computed RDTO. Comparisons of the calculated reactor outputs on a percentage basis are made with the readings and/or the average of the power range nuclear channels. If the computer reactor outputs deviate more than a preset amount from the power range nuclear channel readings, an alarm will be given to alert the plant operators to re-calibrate the power range channels. During steady-state operation this program will also compare the differences between RDTO and RTO and initiate an alarm if the differences exceed a preset amount.

18.0.2.3.3 Steam Generator Total Thermal Output from the Secondary Side (SGTTO)

The heat added to the secondary cycle will be calculated by the increase in enthalpy of the Feedwater supplied to the steam generator. Corrections for the moisture content of the steam generator outlet, for mismatches between steam and feedwater flows, and for blowdown will be made if required. The mismatch measurement error between steam flow and feedwater flow will be determined each time the flow measurement equals the previous one. This measurement error will be used to correct the actual calculated flow mismatch during transient conditions.

18.0.2.3.4 Reactor Thermal Output (RTO)

Since an accurate flow measurement is not feasible on the reactor coolant side of the nuclear steam generator, the reactor thermal output is calculated by using the steam generator total thermal output and adding to it all losses between the reactor and the steam side of the steam generators.

18.0.2.3.5 Reactor Dynamic Thermal Output (RDTO)

This calculation estimates the reactor thermal output at steady- state operation and during transient load changes. The estimate is based on temperature and pressure measurements in the reactor coolant loops during the previous time interval.

18.0.2.3.6 Unit Net Efficiency

The unit net electrical power output is used with the reactor thermal output power to calculate the unit net efficiency. The unit gross output and unit auxiliary kilowatt-hours from contact pulse interrupt inputs will be accumulated by the computer. The kilowatt-hours thus accumulated will be used to calculate the unit net electrical output.

18.0.2.3.7 Tilting Factors

Steam Generator Thermal Output Tilt

The steam generator thermal output tilting factors are calculated by taking the ratio of the thermal output of each steam generator to the average thermal output of all steam generators.

Reactor Coolant Loop ΔT Tilt

The reactor coolant loop ΔT tilting factor is calculated by taking the ratio of the ΔT's in each loop to the average ΔT for all loops.

Radial or x-y Distribution Flux Tilt (Quadrant Power Tilt)

The nuclear power range signals obtained from the upper section of the ionization chambers will be compared with each other. Similar comparisons will be made for the nuclear power range signals derived from the lower sections of the ionization chambers. Existence of a radial flux tilt can be verified by the use of incore instrumentation.

Axial Flux Tilt

The nuclear power range signals from the upper sections of the ionization chambers will be summed and/or averaged. They will then be compared with the sum and/or average of the signals from the lower sections of the ionization chambers, and the operator will be informed with a message of axial tilt deviation between the upper and lower sections.

18.0.2.3.8 Load Follow

This program calculated boron concentration, predicts boration rates necessary, calculates xenon concentration, predicts future xenon concentrations, and performs reactivity calculations.

18.0.2.3.9 On-site In-Core Data Collection

Westinghouse PWR's are equipped with incore instrumentation consisting of thermocouples at the outlet of several fuel assemblies, and vertical flux thimbles in which neutron monitors operate. About one out of three such assemblies contains a flux thimble, and the same is true of thermocouples. Movable miniature fission chambers will be used for incore neutron monitoring, with one detector per group of ten flux thimbles. The computer will output incore data on punched paper tape in a format suitable for teletype transmission.

18.0.2.3.10 Onsite InCore Data Reduction

The philosophy of incore data reduction is to utilize infrequent, detailed offsite analysis to provide a basis for quick interpretation of incore data by the onsite computer. The computer is thus not required to do time-consuming, multi-dimensional, neutron diffusion calculations to arrive at limiting conditions. Instead, the computer and instrumentation are used to alert plant operators when incore parameters deviate from values shown to be safe by prior analysis.

For analysis of incore thermocouple data, the core will be divided into regions. Thermocouple readings (converted to enthalpy rise) will be compared region-wise to check for possible peaking or asymmetry. The variation of this type of data through several thermocouple maps will also be available to the operator so that trends can be identified at an early stage.

Much the same approach will be taken with incore flux monitors. Data from flux scans (corrected for amplifier scale, detector, calibration, etc.) will be processed to give axial power peaking, axial power sharing, and the trend of these with time.

Figure 18.0-8 is a thermocouple output summary. Included on the printout are rod positions, temperature, flows, as well as individual thermocouple temperatures. Relative fuel assembly power is given for each thermocouple location and radial tilting factors I1 through T8.

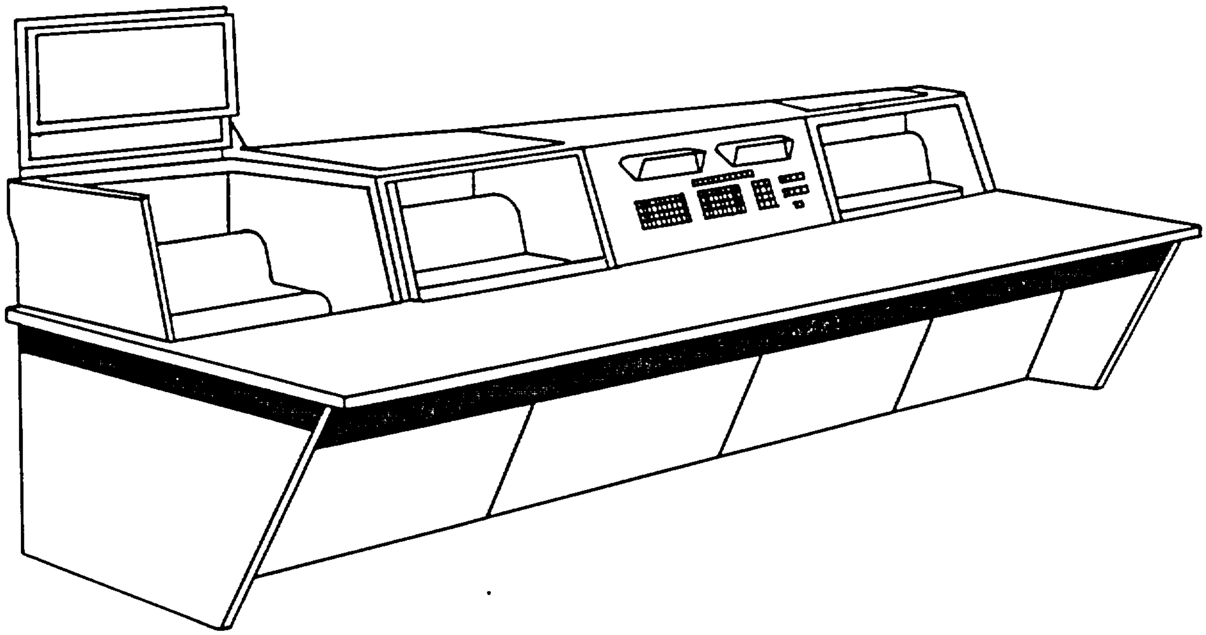
For an understanding of radial tilting factors, refer to Figure 18.0-8, a thermocouple map. Under symmetry check, T1 through T4 are the four quadrants. Then the quadrant divisions are rotated 45 degrees and the quadrants are now designed T5 through T8. At each thermocouple location on the map, the temperatures, delta T, and relative power are recorded.

Figures 18.0-9 shows the incore movable detector data for a single pass of 6 detectors. As the detectors are withdrawn from the top to the bottom of the core at 12 feet per minute, detector voltage is taken each second, which gives 61 core elevations. Figure 18.0-10 is a summary of movable detector data for all passes made by the detectors.

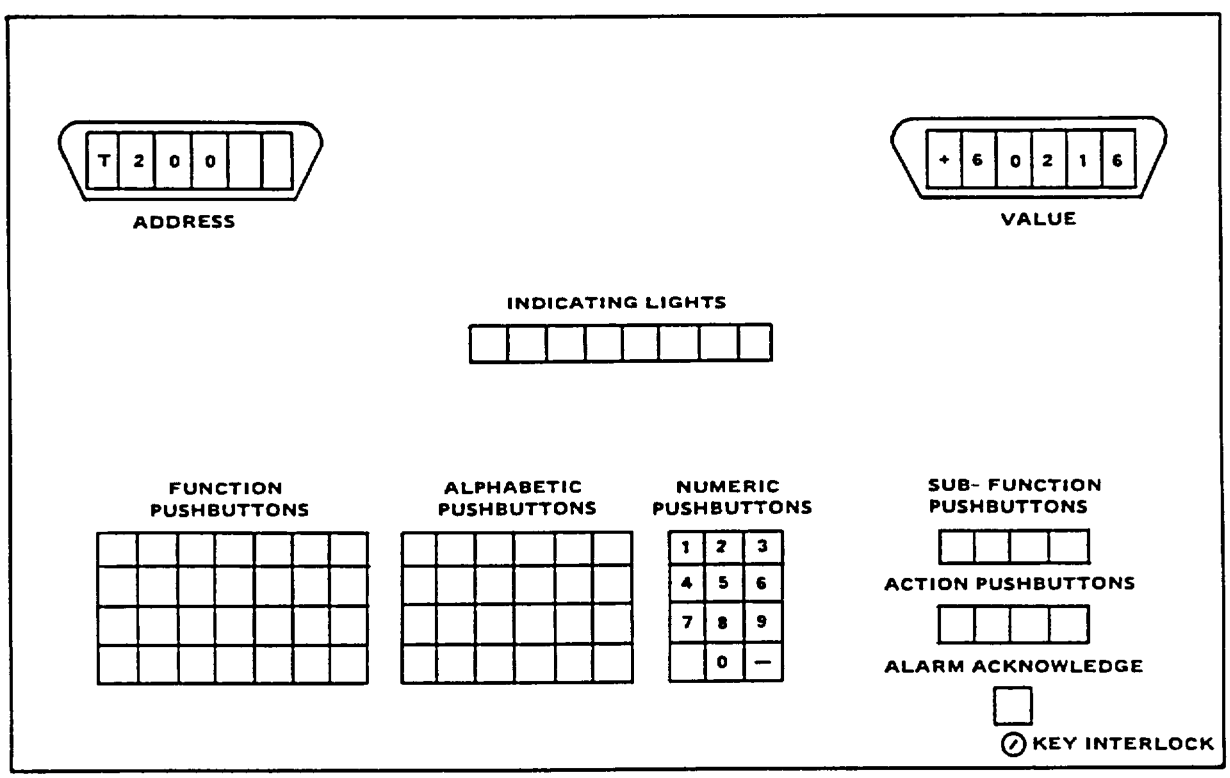
18.0.3 Summary

Although the plant computer is not necessary for plant operation, it is a valuable operational tool. It will monitor the important primary plant analog and digital parameters and store them for future use in various computer programs. Two of the most important programs are the Post Trip Review and the Sequence of Events Record which gives a printout of analog values and digital inputs just before and after a reactor trip to aid in reconstructing the event. The computer also provides for trending selected parameters and alarming off-normal conditions.

The computer will check the operation of various control systems and the reactor protection system. It will do heat balance calculations and provide the calorimetric results used to adjust nuclear instruments. The computer is also used to collect and prepare data on peaking factors for analysis by other computers.



PRODAC OPERATOR'S CONSOLE



CONTROL PANEL

Figure 18—1 Prodac Operators Console

1836 POST TRIP REVIEW

N0049A	N0050A	N0051A	N0052A	TIME 1	P0398A	P0399A	00340A	T049A	TIME 2
95.80	96.45	96.62	96.16	3252 4	422.7	423.1	929.2	572.1	3252.5
95.86	96 56	96 62	96.16	3254.2	113.2	117 2	396.1	553.9	1254.2
95.80	96 68	96 74	96 27	3256 4	92.4	95.0	327.5	552.5	3256.5
56.72	58.01	57.66	51.98	3258 3	73.2	77.6	12.4	551.7	3258.4

1837 POST—TRIP DATA—TRIP TIME 183258

4.69	4.75	4 69	4.63	3300.5	79.8	82 4	4.4	551.9	3330 5
3.81	3 93	3 87	3 70	3302 4	81.5	84.1	—4 0	551.9	3302.4
3.23	3.28	3 23	3 05	3304 4	81.5	83.7	—4.0	552.0	3304 4
2.41	2.46	2.41	2.34	3308 4	80 2	82.8	—4.0	551.8	3308.4

N0031A	N0032A	N0035A	N0036A	N0041A	N0042A	N0043A	N0044A	N0045A	N0046A	N0047A	N0048A	N0049A	N0050A	N0051A
0.03	0.03	305.04	349.11	3.60	4 03	3 61	3.99	3 64	4.03	3 66	4.00	95.92	96 68	96.80
0 03	0.03	299.61	349.11	3 59	4.03	3 60	3 98	3 63	4.02	3 66	4 01	95 80	96.51	96.62
0 03	0 03	296.92	345.98	3.58	4.02	3.60	3 98	3 63	4 01	3.66	4.00	95.57	96.56	96.62
0.03	0 03	296.92	345.98	3 59	4 03	3.61	3.98	3 63	4.01	3.65	3.99	95 80	96.74	96.56
0.03	0 03	302.31	342 88	3.60	4 03	3.59	3.97	3.64	4 03	3.65	4.00	95.86	96 39	96.80
0 03	0.03	299.61	342.88	3.59	4.02	3 60	3.98	3.63	4.02	3.66	4.00	95 62	96.62	96 68
0 03	0 03	299.61	339.81	3 60	4 03	3 60	3 98	3.63	4.01	3.66	3.99	95.80	96.51	96.56
0.03	0 03	305 04	345.98	3.60	4.03	3.61	3.98	3.63	4 02	3.66	4.00	95 86	96 62	96.62
0 03	0 03	305.04	349.11	3 60	4.03	3 60	3.98	3.63	4 02	3 67	4.01	95.80	96 51	96 68
0.03	0 03	305 04	342.88	3.60	4.04	3 61	3.98	3.64	4.02	3.66	4.00	95.92	96.74	96.74
0 03	0.03	299.61	339.81	3.59	4.03	3 60	3.97	3.63	4 02	3.66	4.00	95 69	96 51	96.62
0.03	0 03	302.31	342.88	3.59	4.03	3.61	3 98	3 64	4.02	3.66	4.00	95.69	96 68	96 80
0 03	0.04	304.04	345 98	3.59	4.03	3.60	3.98	3.64	4 02	3.67	4.01	95.80	96.51	96.86
0.03	0 03	299 01	342.88	3.59	4.03	3.60	3 97	3.64	4 02	3 66	4.00	95.74	96.56	96 80
0.03	0 03	299.61	342.88	3.59	4.03	3.60	3.98	3 63	4 02	3.66	4.00	95 80	96.51	96.62
0.03	0 03	299 61	342.88	3 59	4.03	3.60	3 98	3.64	4.02	3 66	4.00	95.69	96 51	96 86

1844 POST—TRIP DATA—TRIP TIME 183258

0 03	0 04	205.35	235.01	2 03	3.27	2.07	3.23	2.08	3.26	1.84	3.08	56.72	58 01	57.66
0.03	0 04	8.20	9.31	0.10	0 11	0.10	0.11	0.11	0.12	0.11	0.10	2.76	2 87	2 81
0 03	0 04	5.05	5.73	0.07	0.07	0.07	0.07	0 08	0.07	0.06	1.76	1.87	1.76	1.76
0 03	0.04	3.68	4.14	0 05	0.05	0 05	0 05	0.05	0 06	0.05	0.04	1.29	1.41	1.35
0 03	0.04	2.79	3.13	0 04	0.03	0.03	0 03	0 04	0.04	0.04	0.02	1.00	1.05	1 00
0 03	0.04	2.05	2 31	0.03	0 02	0 03	0.03	0 03	0.04	0.03	0 01	0.82	0 82	0 82
0 03	0 04	1.65	1.88	0 03	0.02	0 02	0 02	0.03	0 03	0 03	0 01	0 70	0.82	0 77
0.03	0 04	1.32	1.49	0.02	0.02	0 02	0 02	0.02	0 03	0.02	0.00	0.53	0,59	0.53
0 03	0 04	1.14	1.29	0.02	0.01	0.02	0 01	0 02	0.03	0 02	0 00	0.53	0 59	0 53
0 03	0 04	0.95	1 06	0.02	0 01	0 02	0.01	0.02	0 03	0.02	0.00	0 47	0 53	0 47
0.03	0 03	0.76	0.85	0 01	0.00	0 01	0 00	0.01	0 02	0.01	—0.01	0.30	0.35	0 30
0 03	0.03	0.62	0.70	0 01	—0.00	0.00	0.00	0 01	0.01	0.01	—0.01	0.18	0 23	0.23
0 03	0.03	0.53	0.59	0.01	0 00	0.01	0 00	0.01	0.01	0 01	—0 01	0.18	0.30	0.23
0 03	0 03	0 45	0.50	0 01	0.00	0.01	0.00	0 01	0 01	0.01	—0 01	0.23	0 30	0 23
0.03	0 03	0 37	0.41	0 00	—0.00	0 00	0.00	0.01	0 01	0.00	—0.01	0.12	0.23	0.18
0 03	0.03	0 32	0.35	0 00	—0.00	0.00	—0 00	0.00	0 01	0.00	—0.01	0.12	0.18	0.12
0 03	0 03	0.27	0.30	0.00	—0.00	0.00	—0 00	0 00	0 01	0.00	—0 01	0.18	0.18	0.18
0 03	0 03	0.23	0.26	0 00	0 00	0.00	0.00	0 01	0.01	0.01	—0 01	0.18	0 23	0 18
0.03	0 03	0.20	0 22	0 00	—0.01	0 00	—0 00	0 00	0 01	0.00	—0 02	0 06	0.23	0.06
0.03	0 03	0.17	0.19	0 00	—0.01	0 00	—0.00	0 00	0 00	0.00	—0 02	0 00	0.12	0 06
0.03	0 03	0 15	0.16	0 00	—0.01	0 00	—0.00	0.00	0 00	0.00	—0 02	0 00	0.18	0 00
0 03	0.03	0.13	0 14	0 00	—0 00	0 00	—0 00	0.00	0.00	0 00	—0.02	0 00	0.18	0 06
0 03	0.03	0.11	0 13	0 00	—0 01	0 00	—0 00	0.00	0 00	0 00	—0.02	0.00	0.18	0 00
0 03	0 03	0.10	0.11	—0 00	—0 01	0.00	—0 01	—0 00	0.00	—0 00	—0 02	—0.05	0 06	—0 05

Figure 18—2a Post Trip Review

N0052A	TIME 1	P0398A	P0399A	00340A	T0486A	TIME 2	F0403A	F0404A	F0405A	F0406A	F0423A	F0424A	F0425A	F0426A
96 09	3050 2	575.6	580.9	1114 8	578.2	3050.3	3523.6	3519.8	3309.1	3418.3	3514 6	3544.3	3285 6	3279 9
96 16	3058.3	574.7	578 7	1114.1	578.1	3058.3	3610 0	3594.5	3394.7	3482.8	3594.5	3625 4	3270 5	3352.6
96.16	3106.3	571.7	577.8	1114 4	577.8	3106.4	3572.1	3514.6	3302.4	3447.1	3558.3	3580 8	3283 7	3319 2
95 92	3114.3	573.4	577 4	1112.8	577.9	3114.3	3616.8	3585.9	3432.7	3523.3	3599.7	3608.3	3293.1	3287 4
96 04	3122.3	573.4	577.4	1113.2	578.1	3122.3	3568.7	3565.2	3394.7	3475.7	3577.3	3587.7	3232 5	3270 5
96.21	3130.2	573 0	577.8	1113.2	577.9	3130.3	3591.1	3540.8	3454.3	3430 9	3542.6	3579 0	3348 9	3360.0
96 16	3138.3	573 4	577.4	1113 2	578.1	3138.3	3558.3	3540 8	3347.1	3436.3	3560 0	3570 4	3261.0	3302.4
96.16	3146 2	575.2	579.1	1112.8	578.1	3146.2	3584.2	3577.3	3361.8	3434.5	3523.3	3549.6	3380.1	3385 6
96 45	3154.3	575.2	579.6	1116.9	578.1	3154.4	3604 8	3584.2	3463.2	3488.1	3554.8	3587.7	3341.5	3414.7
96 21	3204.4	574.7	579 6	1115.2	578.2	3204 5	3621.9	3560.0	3332.2	3434.5	3599.7	3633.9	3268 6	3308.1
96.16	3210 3	573.0	577.4	1113 6	578.1	3210.3	3603.1	3563.5	3454.3	3521.6	3611.7	3632 2	3259.1	3278 0
96.45	3226.3	576.1	580 4	1114.1	578.2	3226.3	3587.7	3572.1	3425.5	3436.3	3558.3	3577.3	3304.3	3339 7
96.04	3234.2	571.3	575.2	1112.4	577.9	3234.3	3572.1	3547.8	3380.1	3439.9	3549 6	3577.3	3257.2	3304 3
96 27	3242.3	573.4	576.5	1113 2	578.1	3242.3	3585.9	3554.8	3432.7	3482.8	3547.8	3554.8	3304.3	3272.4
96 04	3250.3	573 0	576.5	1112 4	577.9	3250.3	3585.9	3533 9	3391.1	3454.3	3539.1	3558.3	3298.7	3302.4
1859 POST—TRIP DATA—TRIP TIME 183258														
51.98	3258.3	73.2	77 6	12.4	551.7	3258.4	2933.8	2929 6	1669.1	1755.7	2582.0	2582.0	1762.7	1821 4
2.70	3306 2	81.5	84 6	—4.0	552.0	3306.3	2735.4	2692.1	1627.9	1691.2	2880.7	2891.4	1794.0	1838 2
1.64	3314.3	75 0	77.6	—4 0	551.5	3314.4	2477.0	2436.7	892.2	834.9	2659 8	2683 0	1114.0	1108 4
1.29	3322.2	68.9	71.1	—4 0	551.2	3322.3	1759.2	1727.3	581.9	549.2	2504.2	2516.6	613.0	602.8
0.88	3332 5	65.0	68.1	3 6	550.9	3332.5	322.2	205.0	537.8	463.7 *	0.0 *	0.0	581.9	560.3
0.70	3340 4	62.8	65.9	3 6	550 8	3340.5	281.2	172.2	526.2	436 3 *	0.0 *	0.0	537 8	537.8
0 65	3348.4	62 0	63.7	4.0	550.7	3348.5	258.3	131.5	514 3	421.9 *	0 0 *	0 0	537.8	514.3
0.47	3356.4	59.4	61.6	4.0	550.6	3356.5	285.3	131.5	502.1	436.3 *	0.0 *	0 0	526.2	502.1
0 47	3402.3	57.6	60 2	5 2	550.5	3402.3	258 3 *	0.0	514.3	436 3 *	0 0 *	0.C	537 8	526.2
0 41	3410.3	55.9	58.5	4.4	550.5	3418.4	281.2	70.3	514.3	436.3 *	0 0 *	0.0	537.8	514.3
0.18	3418.3	52.9	55.9	5.6	550.2	3418.4	172.2 *	0.0	489.7	391.5 *	0 0 *	0 0	502.1	489.7
0.12	3426.2	50.7	53.7	3.2	550.1	3426.3	172.2 *	0 0	476 9	375.4 *	0 0 *	0.0	489.7	476.9
0.18	3434.3	50 2	52 4	5 6	550.0	3434.4	172.2 *	0 0	489.7	391.5 *	0 0 *	0.0	514.3	476 9
0.18	3442.3	49.4	52.4	4.8	550.0	3442.3	205.0 *	0.0	489.7	391.5 *	0.0 *	0 0	514.3	489.7
0.06	3450.3	47.6	50.2	2.0	549.9	3450.4	172.2 *	0 0	476.9	375.4 *	0 0 *	0 0	489.7	463.7
0 06	3458.2	46.3	48.9	0 4	549.8	3458.3	172.2 *	0 0	489.7	375 4 *	0 0 *	0 0	502.1	502.1
0.06	3506.2	45.0	47 6	0.0	549.7	3506.3	172 2 *	0 0	489.7	391.5 *	0 0 *	0.0	514.3	476 9
0.12	3514.3	43.7	46.3	0.8	549.6	3514.3	205.0 *	0.0	489.7	391.5 *	0.0 *	0.0	514.3	502.1
0 00	3522.3	42 0	45.4	1.2	549.6	3522.3	131.5 *	0 0	463.7	375.4 *	0.0 *	0 0	489.7	463.7
—0.05	3530.2	40.7	43.7	0.8	549.4	3530.3	131.5 *	0 0	463.7	375 4	0.0 *	0 0	[illegible]	463.7
—0 05	3538.3	39.4	42.9	0.8	549.4	3538.4	131.5 *	0.0	476.9	391.5 *	0.0 *	0.0	489.7	476 9
—0.05	3546.2	37.6	41.6	0.8	549.3	3546.3	172.2 *	0.0	476.9	407 0 *	0.0 *	0 0	502.1	489.7
0.00	3554.3	37.2	39.8	0.4	549.2	3554.4	70.3 *	0.0	463.7	375.4 *	0.0 *	0 0	476.9	450.2
—0.12	3602.2	35 0	38.1	0.4	549.2	3602.3 *	[illegible]	0 0	450.2	340.9 *	0.0 *	0.0	463.7	436.3
1913 POST—TRIP DATA—TRIP TIME 183238														
2645.8	2577.2	1780.2	1868.2	2676.0	[illegible]22.1	1577.7	1593.3	3256.6	957.6	952.5	964.1	951.0	953.1	960.5
3008.7	2983 9	1538.1	1612.6	2857.0	2992.2	1521.9	1517.8	3304.6	973.0	968.5	979.5	967.9	970.7	976.6

1914 POST TRIP REVIEW STOPPED

1914 POST TRIP REVIEW FINISHED

Figure 18—2b Post Trip Review

1833		SEQUENCE OF EVENTS RECORD. FIRST EVENT AT H18 M32 S57			
	L0406D	STM GEN A LO LO L CAUS RE	TR	C	0
	L0406D	STM GEN A LO LO L CAUS RE	NT TR	C	1
	L0406D	STM GEN A LO LO L CAUS RE	TR	C	1
	L0466D	STM GEN D LO LO L CAUS RE	TR	C	4
	Y0006D	REAC MAIN TR BKR A	TR	C	6
	Y0007D	REAC MAIN TR BKR B	TR	C	7
	Y2008D	TB TRIP — SOL ENERGIZED	TR	C	16
	Y0390D	TB STOP VALVES CL & P7 CAUS RE	TR	C	16
	N0025D	PWR RNG CHAN 1 HI Q RATE PART RE	TR	C	26
	N0029D	PWR RNG CHAN HI Q RATE CAUS RE	TR	C	26
	N0026D	PWR RNG CHAN 2 HI Q RATE PART RE	TR	C	26
	N0027D	PWR RNG CHAN 3 HI Q RATE PART RE	TR	C	26
	N0028D	PWR RNG CHAN 4 HI Q RATE PART RE	TR	C	26
	Y2011D	TB TRIP — EH CONTROL PWR	TRBL	C	42
	L0446D	STM GEN C LO LO L CAUS RE	TR	C	46
	L0426D	STM GEN B LO LO L CAUS RE	TR	C	50
	N0006D	PWR RNG CHAN 1 LO Q PART RE	NT TR	C	57
	N0007D	PWR RNG CHAN 2 LO Q PART RE	NT TR	C	57
	N0008D	PWR RNG CHAN 3 LO Q PART RE	NT TR	C	57
	N0009D	PWR RNG CHAN 4 LO Q PART RE	NT TR	C	57
	N0021D	INTERM RNG 2 HI Q INITIATES RE	NT TR	C	61
	N0020D	INTERM RNG 1 HI Q INITIATES RE	NT TR	C	61
	P0488D	PRESSURIZER LO P & P7 CAUS RE	TR	C	150
	L0406D	STM GEN A LO LO L CAUS RE	NT TR	C	176
	L0466D	STM GEN D LO LO L CAUS RE	NT TR	C	243
	L0446D	STM GEN C LO LO L CAUS RE	NT TR	C	247
	L0466D	STM GEN D LO LO L CAUS RE	TR	C	292
	P0488D	PRESSURIZER LO P & P7 CAUS RE	NT TR	C	305
	L0426D	STM GEN B LO LO L CAUS RE	NT TR	C	307
1835		END SEQUENCE OF EVENTS RECORD			

Figure 18—3 Sequence of Events

	LOG 1	UNIT												
					LOAD								T REF	
	TIME	NET GENERATION	AUX CONSUMPTION	GROSS GENERATION	GROSS	AUX	NET	STEAM GEN THREMAL OUTPUT	REACTOR THREMAL OUTPUT	UNIT NET EFF	NUCLEAR POWER	AUCT T AVG	MEAS	DEV FROM COMP
		MWH	MWH	MWH	MW	MW	MW	MWH	MWH	%	%	°F	°F	
	HR MIN	SINCE LAST HOUR			LAST ONE MINUTE AVG.			SINCE LAST HOUR			LAST ONE MINUTE AVERAGES			
	POINT	U0380	———	U0313	Q0340A	———	———		U1118	U1120	U1150	T0499A	T0496A	U0487
1	0010	140 0	8 0	148.0	889.9	0.0	12.0	2822 6	2811 0	30 6	81.5	572	571	−0.2
2	0100	864 0	36 0	900 0	898.2	0.0	20 0	2829 0	2817.3	30.7	82.1	572	572	−0.2
3	0200	868.0	36.0	904.0	904.9	0.0	16 0	2852 7	2841.0	30 6	83 0	573	572	−0 2
4	0300	884 0	36.0	920.0	923 0	0 0	16 0	2894.1	2882.4	30.7	84 6	574	572	0 2
5	0400	888.0	36.0	924 0	921.6	0 0	12 0	2912 6	2900.9	30.6	84 4	574	572	−0.2
6	0500	896 0	36.0	928.0	924.8	4 0	16 0	2914.0	2902 3	30.9	84 5	574	572	−0 1
7	0600	888.0	36 0	924.0	919.3	0.0	16 0	2901.2	2889.5	30.7	84.3	575	572	−0 3
8	0700	888 0	32.0	920 0	916 6	0 0	16 0	2894.3	2882 6	30.8	83 3	573	572	−0.4
9	0800	884 0	40 0	924 0	924.3	0.0	12 0	2907 7	2896.1	30 5	85.1	575	572	−0 1
10	0900	900.0	32.0	932.0	930 6	0 0	16 0	2916.5	2904.8	31.0	85 9	576	572	−0.3
11	1000	932.0	40 0	972 0	1006.1	0 0	20 0	3047.7	3036.0	30.7	92.7	576	576	−0 2
12	1100	1016 0	36 0	1052.0	1072.9	0.0	12 0	3323.3	3311.6	30 7	99 3	578	578	−0 4
13	1200	1052 0	32.0	1084.0	1076 7	0.0	20 0	3407.6	3395.9	31.0	99.8	579	578	−0.5
14														

Figure 18—4a Periodic Log

0986

	LOG 2	REACTOR INCORE																				
	TIME	HOTTEST THERMO-COUPLE		AVG OF T/C	QUADRANT FUEL ASSEMBLY POWER TILT								ROD BANK POSITIONS									
					1 | 2 / 4 | 3				5 / 8 X 6 / 7				CONTROL				SHUTDOWN				PART LGTH A
		LOC	VALUE °F	°F	1	2	3	4	5	6	7	8	A	B	C	D	A	B	C	D	
	HR MIN	INSTANTANEOUS											STEPS				STEPS				
	POINT		U0090	U0091	U1159	U1160	U1161	U1162	U1151	U1152	U1153	U1154	U0049	U0050	U0051	U0052	U0053	U0054	U0055	U0056	U0057
1	0010	14	605 4	595 1	1 00	1.00	1 01	0 99	1 01	1.01	0 99	0 99	228	228	228	200	228	228	228	228	1 *
2	0100	14	605 2	594 7	1 00	1 01	1.01	0 99	1 01	1 01	0 99	0 99	228	228	228	200	228	228	228	228	1 *
3	0200	14	606 6	596 2	1.00	1 00	1.01	0 99	1.01	1 01	0.99	0 99	228	228	228	200	228	228	228	228	1 *
4	0300	14	607 9	597.2	1 00	1.00	1 01	0 99	1 01	1 01	0 99	0 99	228	228	228	200	228	228	228	228	1 *
5	0400	51	608 1	597.2	1 00	1.00	1 01	0 99	1 01	1.01	0 99	0 99	228	228	228	200	228	228	228	228	1 *
6	0500	14	607.7	597 2	1 00	1 14	1.10	0 76	1 01	1 12	0 88	0 98	278	228	228	200	228	228	228	228	1 *
8	0700	14	606 6	596 1	1 00	1 05	1 03	0 92	1 02	1.04	0 96	0 98	228	228	228	202	228	228	228	228	1 *
9	0800	14	609 2	589 3	0 85	0 60	1 34	1.21	−0 09	1.33	1 50	1.27	228	228	228	202	228	228	228	228	1 *
10	0900	14	610.2	590 4	1 07	0 56	1 20	1 18	0.23	1 23	1.34	1 20	228	228	228	200	228	228	228	228	1 *
11	1000	14	613 2	602.0	0 55	0 98	1 27	1.20	[illegible]	1 19	[illegible]	0 88	228	228	228	209	228	228	228	228	1 *
12	1100	51	618.0	605 5	0 90	1.00	1.06	1 04	0 87	1 04	1 08	1 02	228	228	228	212	228	228	228	228	1 *
13	1200	14	618 6	597.5	0 91	1 00	1 05	1.04	0 88	1.04	1 07	1 01	228	228	228	209	228	228	228	228	1 *
14																					

Figure 18—4b Periodic Log

GENERATOR									REACTOR INCORE								
GEN H_2	H_2 COLD GAS TEMP		H_2 SEAL OIL TEMP		GENERATOR		EXCITER AIR		CALC BORON CONC IN COOL-ANT	XENON WORTH	SHUT DOWN MARGIN	HOT PH FACTOR	PRESSURIZER				
	NO. 1	NO 2	AIR SIDE	GAS SIDE	VARS	STA WDG (HIGH-EST)	IN LET (AVG)	OUT LET (AVG)					PRESS	LEVEL	LEVEL SET POINT	LVL SP DEV FROM COMP	STEAM TEMP
PSIG	°F		°F		MVAR	°F	°F		PPM	PCM	PCM		PSIG		%	%	°F
LAST ONE MINUTE AVERAGES																	
P2800A	T2830A	T2831A	T2810A	T2811A	Q2823A	T2800A 5A	T2812A 13A	T2814A 15A	U1300	U1600	U1601	U1311	U0482	U0483	L0483A	U0488	T0481A
70	103	103	96	105	126	114 *	100	98	697 *		.0	3 20	2235	55 7	53 7	3 5	650 3
70	103	104	95	106	136	144 *	99	98	697 *		.0	3 20	2229	55 4	53 3	3 5	650 7
70	103	104	96	106	135	115 *	100	98	697 *		.0	3 20	2232	56 4	54 5	3 8	650 7
69	103	104	95	105	124	116 *	100	98	698 *		.0	3.20	2234	57 4	55 5	3 8	649 9
69	104	104	96	105	121	116 *	100	98	699 *		.0	3 20	2231	57 2	55 4	3 7	650 0
69	104	104	95	105	118	116 *	100	98	700 *			3 21	2235	57 4	55 8	3 9	650 7
69	103	104	95	105	118	116 *	99	98	701 *		.0	3 20	2231	56 3	53 8	3 6	650 9
69	103	104	95	105	123	115 *	100	98			.0						650.1
69	103	104	95	106	129	115 *	100	98	700 *		.0	3 21	2229	58 7	57 0	3 9	650 2
69	103	104	95	105	108	116 *	100	97	700 *		.0	3 21	2232	59 6	57 3	4 1	649 3
70	104	105	96	107	1/3	119 *	100	98	700 *		.0	3 21	2234	59 7	5/ 8	4 0	649.1
71	107	107	96	108	20/	124 *	101	99	700 *		.0	3 22	2233	61 6	60 0	4 3	650 4
72	108	107	96	109	225	125 *	101	99	707 *		.0	3 22	2232	62 9	60 9	5 0	650.3

Figure 18—4c Periodic Log

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EXAMPLE 1: (ALARM TYPEWRITER)

000019 C0022A CONT ROD BNK D GROUP 1 POS B10 DEV FROM BANK —16 STEPS

EXAMPLE 2: (ALARM TYPEWRITER)

1015 REDUNDANT MEASUREMENT STATUS REPORT

C0021A	CONT ROD BNK D GROUP 1 POS F02	143.00
C0022A	CONT ROD BNK D GROUP 1 POS B10	Q 130.00
C0023A	CONT ROD BNK D GROUP 1 POS K14	143.00
C0024A	CONT ROD BNK D GROUP 1 POS P06	143.00
C0025A	CONT ROD BNK D GROUP 2 POS B06	143.00
C0026A	CONT ROD BNK D GROUP 2 POS F14	143.00
C0027A	CONT ROD BNK D GROUP 2 POS P10	143.00
C0028A	CONT ROD BNK D GROUP 2 POS K02	143.00
C0029A	CONT ROD BNK D GROUP 2 POS H08	143.00
K0014	CONT & SD ROD NOMAL CLUSTER LIM	10.00

EXAMPLE 3: (ALARM TYPEWRITER) (IN RED INK)

120153 ROD BANK SEQUENCE

U0053	SD ROD BANK A	0 STEPS FRM TOP
U0054	SD ROD BANK B	0 STEPS FRM TOP
U0055	SD ROD BANK C	230 STEPS FRM TOP
U0056	SD ROD BANK D	210 STEPS FRM TOP

EXAMPLE 3A:

112057 ROD BANK SEQUENCE

U0049	CONTROL ROD BANK A AT 230 STEPS
U0050	CONTROL ROD BANK B AT 230 STEPS
U0051	CONTROL ROD BANK C AT 220 STEPS
U0052	CONTROL ROD BANK D AT 100 STEPS

Figure 18—5a Rod Position Supervision Output

EXAMPLE 4: (ALARM TYPEWRITER)

000635 C0022A CONT ROD BNK D GROUP 1 POS B10 DEV FROM BANK —1STEPS RETURN TO NORMAL

111737 ROD BANK SEQUENCE

U0049	CONTROL ROD BANK A AT 230 STEPS
U0050	CONTROL ROD BANK B AT 230 STEPS
U0051	CONTROL ROD BANK C AT 230 STEPS
U0052	CONTROL ROD BANK D AT 230 STEPS

1013 REDUNDANT MEASUREMENT STATUS REPORT

C0021A	CONT ROD BNK D GROUP 1 POS F02	143.00
C0022A	CONT ROD BNK D GROUP 1 POS B10	143.00
C0023A	CONT ROD BNK D GROUP 1 POS K14	143.00
C0024A	CONT ROD BNK D GROUP 1 POS P06	143.00
C0025A	CONT ROD BNK D GROUP 2 POS B06	143.00
C0026A	CONT ROD BNK D GROUP 2 POS F14	143.00
C0027A	CONT ROD BNK D GROUP 2 POS P10	143.00
C0028A	CONT ROD BNK D GROUP 2 POS K02	143.00
C0029A	CONT ROD BNK D GROUP 2 POS H08	143.00
K0014	CONT & SD ROD NORMAL CLUSTER LIM	10.00

Figure 18—5b Rod Position Supervision Output

EXAMPLE 1: (TREND TYPEWRITER)

```
0129   REAC PROT SYST—GROUP TRIP STATUS
       HIGH SOURCE RANGE FLUX
       TRIPS OPERATED
       N0031D     TR          26248
       STATUS OF TRIPS & PREM THAT INIT DATA COLL.
       N0030D     NT TR
       N0031D     TR
       STSTUS OF TRIPS & PERM THAT DO NOT INIT DATA COLL.
       N0034D     SET
       N0035D     RESET

       Y0006D     TR
       Y0007D     TR
       Y0026D     TR
       Y0027D     TR
       STATUS OF VARIABLES & SET POINTS
       K0083         400.00     N0031A     398.10     26269
       K0084         400.00     N0032A     398.10     26269
RPSM   GROUP 1 COMPLETE
```

EXAMPLE 2: (ALARM TYPEWRITER)

```
0130   RETURN TO NORMAL STATUS—HIGH SOURCE RANGE FLUX
       N0031D     NT TR       26404
```

EXAMPLE 3: (ALARM TYPEWRITER)

```
0150   RPSM DATA STORAGE FULL—BUS TURBINE TRIP PERMISSIVE
```

Figure 18—6 RPSM Output

1745 DATE: 11/3/81 TIME START 7.43.35 UP TO 7.44.32
POWER 3366.7 MWT 1115.6 MWE
BORON CONCENTRATION 1080.0 PPM

ROD POSITION:

S—GRA	S—GRB	S—GRC	S—GRD	C—GRA	C—GRB	C—GRC	C—GRD
230	228	228	228	228	228	228	211

REACTOR LOOP A VARIABLES:

T—IN	TAVG1	TDT1	FLOW1	FLOW2	FLOW3
550.4	578.3	96.5	101.6	100.7	100.8

REACTOR LOOP B VARIABLES:

T—IN	TAVG1	TDT1	FLOW1	FLOW2	FLOW3
550.1	557.3	98.9	100.3	99.9	100.3

REACTOR LOOP C VARIABLES:

T—IN	TAVG1	TDT1	FLOW1	FLOW2	FLOW3
550.9	557.7	96.4	100.9	100.7	100.0

REACTOR LOOP D VARIABLES:

T—IN	TAVG1	TDT1	FLOW1	FLOW2	FLOW3
550.4	576.5	96.5	100.2	100.4	98.8

HOT JUNCT. BOX A AVG. TEMP. = 160.2
HOT JUNCT. BOX B AVG. TEMP. = 160.6
SYSTEM PRESSURE = 2251.9
AVG. CORE INLET TEMP. FOR THIS CASE = 550.5
AVG. CORE OUTLET TEMP. FOR THIS CASE = 604.0
BYPASS FRACTION = 0.07
AVERAGE INCORE T/C TEMPERATURE: 604.0
INDIVIDUAL INCORE T/C TEMPERATURES (IN ASCENDING ORDER):

589.9		607.4	0.0	613.6	609.4	603.6	615.9	615.6	594.3	608.8
614.2		617.3	609.1	607.8		0.0	615.4	608.7	603.8	616.1
612.1	615.9	604.2	608.8	613.5	612.5	608.0	582.4	588.9	591.9	587.9
581.4	603.4	609.5	0.0	581.1	0.0	615.8	614.8	609.5		617.3
616.6	614.3	582.7	592.2	611.8	0.0	617.9	591.4	589.4	616.7	0.0
614.2	590.4	608.6		550.7	609.6	582.6	602.1	608.1	609.6	

INDIVIDUAL RELATIVE FUEL ASSEMBLY POWERS (IN ASCENDING ORDER):

0.713		[illegible]	0.000	1.220	1.034	0.990	1.268	1.216	0.805	1.108
1.203		1.302	1.128	[illegible]		0.000	[illegible]	1.108	[illegible]	1.280
1.134	1.265	0.998	1.081	1.204	1.199	1.078	0.536	0.701	0.767	0.704
0.557	1.013	1.147	0.000	0.563	0.000	1.315	1.274	1.101		1.280
1.301	1.222	0.585	0.777	1.210	0.000	1.283	0.769	0.710	1.300	0.000
1.163	1.700	1.097		0.004	1.093	0.555	0.941	1.124	1.129	

RADIAL SYMMETRY TILT FACTORS:

T1	T2	T3	T4	T5	T6	T7	T8
1.015	0.995	0.954	1.036	1.041	1.015	0.912	1.032

Figure 18—7 Thermocouple and Fuel Assembly Power

DATE — 11/3/81 TIME — 7.32.55 TO 7 33 52 POWER — 1114 7 MWE

ROD BANK POSITION	GROUP S—A	GROUP S—B	GROUP S—C	GROUP S—D	GROUP C—A	GROUP C—B	GROUP C—C	GROUP C—D
	230 STEPS	228 STEPS	228 STEPS	228 STEPS	228 STEPS	228 STEPS	228 STEPS	211 STEPS

270 DEG. CLOCKWISE

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A					589 9 39.38 0.713										
B		581 5 30.95 0 559		603 4 52.87 1 014				609 2 58 64 1.143		0.0 0 00 0 000		581.1 30 55 0 562			
C			607 4 56.79 1.100		0 0 0 00 0.000		613 6 63 07 1.222				609.3 58 73 1.033				
D		0 0 0 00 0 000				615 9 65 29 1.317				614.8 64.26 1 276		609 3 58 70 1 097			
E	603 7 53 11 0 992				615.9 65.38 1.270				615.3 64.78 1.212				594 4 43 80 0.807		
F				617.5 66 89 1 284				616.3 65.78 1 296				614 4 63.80 1.225		582 7 32 14 0.583	
G	608 9 58.30 1 110		613 8 63.23 1 197								617 4 66.80 1.304		608 9 58.34 1.124		

Figure 18—8a Incore Thermocouple Map

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Figure 18—8b Incore Thermocouple Map

H ***	592 2 41.63 0.776		611.9 61.29 1.212				0 0 0.00 0 000				617 6 67.09 1.279				591.4 40.84 0 770
J ***		607.9 57.34 1.119						0.0 0 00 0 000		615 4 64.82 1.269				608 9 58 31 1.112	
K ***	589.4 38.80 0 709				616 5 65.89 1 296				0.0 0 00 0 000				614 2 63.63 1.164		590 2 39 67 0.698
L ***		603 5 52.89 1.004		616.2 65 67 1.284				612.2 61.60 1.136				615 8 65.20 1.263		604 5 53 89 1 003	
M ***			608 6 58 02 1 097								550.6 0 00 —0.000		609 1 58.55 1 085		
N ***		582 6 32.02 0.554		608 9 58.31 1 082		613.2 62.63 1.198				612.7 62 17 1.205		608.2 57.60 1.081		582 4 31.84 0 536	
P ***					602 0 51 45 0 940		608 2 57.60 1 081		609.1 58.55 1 085						
R ***						589.1 38.49 0.704		592 0 41.40 0.769		587.9 37.35 0 704					

	LOOP—1	LOOP—2	LOOP—3	LOOP—4
INLET TEMP. (RTD)··	550 4	550 1	551.3	550.4
AVG TEMP. (RTD)—	578 2	577 2	577.5	576.4
TEMP. RISE (RTD)—	96 6	98.5	96 4	96 5
FLOW (PERCENT)—	100 7	100 4	100.2	99.5

SYSTEM PRESSURE= 2250.3 SIA
AVG. OUTLET TEMP. FROM INCORE T/C = 604.
AVG. CORE INLET TEMP FOR THIS CASE = 550.6
AVG CORE OUTLET TEMP. FOR THIS CASE = 604 0
THERMAL POWER = 3378 5 MWT
MAIN COOLANT BORON CONCENTRATION = 1080 0 PPM
HIGHEST RELATIVE FUEL ASSEMBLY POWER IS 1.317
AT LOCATION D—7

SYMMETRY CHECK

—1 016 * 0 994
* * * * * * * * * *
1 035 * 0 955

* 1 040 *
1.033 * 1 014
* 0 914 *

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INCORE MOVEABLE DETECTOR DATA

MEAS I.D. .... TIME 9:53 DATE 8/20/81 PASS 12 POWER 3397.8 MWT 1111.5 MWE

BORON CONCENTRATION 688.48 P(NI) 3.73 AD(NI) -2,520

ROD BANK POSITIONS

ROD BANK·	SD A	SD B	SD C	SD D	CONT A	CONT B	CONT C	CONT D
STEP 6:	228	228	228	228	228	228	228	216

DETECTOR CONFIGURATION

DETECTOR·	MD—1	MD—2	MD—3	MD—4	MD—5	MD—6
THIMBLE.	L—5	D—10	G—9	J—7	K—12	J—10
SCALE	30	30	30	30	30	30
LEAKAGE	0 000	0.000	0 000	0 000	0 000	0.000

UN—NORMALIZED VOLTAGES AND INTEGRALS

POINT	VOLTS	INTEGRAL	VOLTS	INTEGRAL	VOLTS	INTEGRAL	VOLTS	INTEGRAL	VOLTS	INTEGRAL	VOLTS	INTEGRAL
1	0 146	0.000	0 200	0.000	0.197	0 000	0 129	0.000	0.132	0 000	0 156	0 000
2	0 174	0.160	0.221	0 210	0.252	0.225	0 167	0.148	0.172	0 152	0 144	0 150
3	0 228	0 360	0.288	0.465	0 317	0.509	0 215	0.339	0 217	0 346	0 188	0.316
4	0.277	0 613	0 351	0.785	0.373	0 854	0.256	0.575	0 256	0.583	0 233	0 527
5	0.321	0.911	0 407	1.164	0.422	1 252	0.292	0.849	0 291	0 856	0.274	0 781
6	0 358	1.250	0 454	1 595	0.454	1.688	0 317	1.153	0 313	1.159	0.310	1 072
7	0 361	1.609	0.458	2.051	0 478	2.151	0 321	1 472	0.321	1.476	0.323	1 388
8	0 417	1.998	0.525	2.542	0.550	2.665	0 380	1.823	0.377	1 825	0.360	1.730
9	0 459	2.436	0.584	3 097	0 588	3.234	0 409	2 217	0.406	2.216	0 409	2.114
10	0 488	2.910	0 620	3.699	0 618	3.838	0 433	2 638	0 428	2.633	0 438	2 537

Figure 18—9a Incore Movable Detector Map

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51	0 560	27 292	0.710	34 539	0 688	33.473	0 478	23.690	0 478	23 460	0 514	25 404
52	0 557	27.851	0.706	35.247	0 648	34.131	0 465	24.161	0 465	23.931	0 507	25 914
53	0 539	28.398	0 681	35.940	0.621	34.766	0 446	24.616	0 445	24 386	0 486	26 410
54	0 513	28.924	0 647	36 604	0 583	35.368	0.421	25 049	0 419	24 818	0.461	26 884
55	0 481	29 422	0 606	37 231	0.535	35.927	0.387	25 453	0 388	25.221	0 427	27 328
56	0 442	29 883	0.559	37 813	0.479	36.434	0 347	25.820	0 347	25.589	0 387	27.735
57	0 396	30.302	0 501	38 343	0.414	36.881	0 302	26.145	0 303	25.914	0 342	28 099
58	0.342	30.671	0 434	38.811	0 341	37.258	0.250	26 421	0.251	26 191	0 291	28 416
59	0 275	30 980	0.349	39.203	0.246	37.551	0 186	26 639	0.184	26 408	0 230	28 676
60	0 199	31.217	0 254	39.505	0.226	37.787	0 148	26.805	0 159	26.580	0 171	28 876
61	0 203	31.418	0 260	39.762	0.165	37.983	0.118	26.939	0.119	26 719	0 218	29 071
FINAL VALUES	61	0.524	61	0.663	61	0 633	61	0 449	61	0 445	61	0 485

BAD DATA INFORMATION

PT VOLTS	PT VOLTS	PT VOLTS	PT VOLTS	PT VOLTS	PT VOLTS

CALCULATED VALUES

TOTAL (NORM)	15 746	19 928	19.037	13.502	13 391	14.570
I (TOP) (NORM)	7.575	9.601	9 432	6 648	6.563	6 992
I (BOT) (NORM)	8.172	10 327	9.605	6 853	6 829	7 579
AO (MD) (NORM)	−3 793	−3.641	−0 906	−1.518	−1 984	−4 030
PEAK (NORM)	19.147	24 007	23.273	16 556	16 306	18 105
PEAK (KW/FT)	19 147	24.007	23 273	16.556	16 306	18 105
FZ	1.216	1.205	1 223	1.226	1 218	1 243

Figure 18–9b Incore Movable Detector Map

Figure 18—10a Summary of Movable Detector Data

MEAS I.D. .... DATE: 8/20/81

NO. OF PASSES 12

POWER 3403 8 MWT

BORON CONCENTRATION 689 50

ROD BANK POSITIONS

ROD BANK	SD A	SD B	SD C	SD D	CONT A	CONT B	CONT C	CONT D
STEPS	228	228	228	228	228	228	228	21(

DETECTOR	THIMBLE	SCALE	TIME	PASS	NORM INTEGRAL	INITIAL POWER	NORM. PEAK	FZ	AD%	PEAK (KW/FT)
MD—1	J—10	30	8 10	1	12 840	3.717	15.798	1 230	—5 080	15 798
MD—2	D—10	30	8·10	1	19.875	3./17	23 848	1.200	—3 757	23 848
MD -3	R-1'	30	8:10	1	9.258	3.717	11 272	1.217	- 1 685	11 272
MD—4	J—7	30	8.10	1	13 454	3 717	16 406	1.219	- 1 529	16 406
MD—5	K—12	30	8 10	1	13.375	3 717	16 223	1.213	—1 842	16 223
MD—6	H—2	30	8.10	1	15 034	1 717	18 281	1 216	—3 725	18 281
.	H—15	30	d:22	2	/.416	3.715	9 020	1.216	-3 113	9 020
MD—2	J—10	30	8.22	2	16 440	3 715	20 354	1 238	2 240	20 354
MD—3	G—9	30	8·22	2	18.981	3.715	23.146	1 219	0 430	23 146
MD—4	R—11	10	8 22	2	5 089	3.715	6 160	1 210	0 453	6 160
MD—5	F—14	30	8·22	2	10 968	3 715	13 203	1 204	1 117	13 203
MD -6	B—3	10	8·22	2	6.111	3 715	7 442	1 218	1 519	7 442

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MD-1	H-3	30	9·44	11	11.786	3 738	14 498	1 230	4 621	14 498
MD-2	E-11	30	9 44	11	20 051	3 738	24,399	1,217	3 014	24 399
MD 3	E-5	30	9·44	11	19 075	3 738	23,185	1 215	1 885	23 185
MD 4	L-11	30	9 44	11	13 500	3 738	16 393	1 214	3 068	16 393
MD 5	J- 10	30	9 44	11	11.07	3 738	13 751	1 241	1 793	13 751
MD-6	H ·2	30	9 44	11	15.010	˙38	18 325	1 221	-[illegible]	18 325
MD-1	L-5	30	9 53	12	15.746	3 726	19 147	1 216	3 793	19 147
MD-2	D-10	30	9 53	12	19 928	3 726	24 007	1 205	3 641	24 007
MD-3	G-9	30	9 53	12	19 037	3 726	23 273	1 223	0 906	23 273
MD-4	J-7	30	9 53	12	13 502	3 726	16 556	1 226	1 518	16 556
MD-5	K-12	30	9.53	12	13 391	3 726	16 306	1 218	1 984	16 306
MD-6	J 10	30	9 53	12	14 570	3 726	18 105	1 243	4 030	18 105
AVERAGES					12.780		15 629	1 223	2 576	15 629

DETECTOR NORMALIZATION FACTORS

DETECTOR	STORED	CALCULATED
MD 1	1 000	1 000
MD 2	1 000	0 781
MD 3	1 000	0 808
MD 4	1 000	1 150
MD 5	000	1 162
MD 6	1 000	0 884

POWER NORMALIZATION FACTORS

PASS	FACTOR	PASS	FACTOR
1	1 005	7	1 000
2	1 005	8	0 999
3	1 004	9	1 002
4	1 001	10	1 001
5	1 000	11	0 999
6	1 002	12	1 002

Figure 18—10b Summary of Movable Detector Data

Westinghouse Technology Systems Manual

Chapter 19

PLANT OPERATIONS

Westinghouse Technology Systems Manual

Section 19.0

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19.0 PLANT OPERATIONS

Learning Objectives:

1. Given a list of plant evolutions similar to the following, arrange the list in the order that they are performed during plant start-up, shutdown, or power operations.

   a. Reactor coolant system (RCS) fill and vent
   b. Establish pressurizer steam bubble
   c. Start reactor coolant pumps
   d. Place all engineered safety features (ESF) systems in an operable condition
   e. Establish no-load T_{avg}
   f. Take the reactor critical
   g. Place main feedwater in service
   h. Synchronize and load the turbine generator
   i. Place rod control system in auto, place steam generator level control in auto, shift steam dump system to T_{avg} mode
   j. Escalate turbine generator load to desired value.

19.1 Introduction

The purpose of this chapter is as follows:

1. Review control, instrumentation and plant systems.

2. Describe plant operations and systems' alignment during shutdown conditions, plant start-up, and power operations.

Basic procedures for start-up, and power operation, of a pressurized water reactor are described in this chapter. This discussion is general in nature and is designed to review the systems, discussed in the previous chapters of this manual, and describe how they are utilized during plant operations. An abbreviated plant start-up procedure is provided in Appendix 19-1 and is used in conjunction with the material contained in this chapter.

19.2 Plant Heatup

19.2.1 Initial Conditions- Reactor Subcritical Plant is Shutdown and in Solid Pressure Control

For the purpose of this discussion, it's assumed that the reactor coolant system has been filled and vented; the reactor coolant pumps (RCPs) are secured, and the plant is aligned and in a solid water condition as shown in Figure 19-1. Solid plant operation is one method of pressure control during a shutdown.

Utilizing this pressure control scheme, the pressurizer and the RCS are completely filled with water. The pressurizer power operated relief valves (PORVs); their associated block valves and the pressurizer spray valves are shut. The pressure in the RCS is controlled between 320 and 400 psig by maintaining a flow balance between the coolant removed from the RCS and the coolant being returned. Coolant removal is accomplished by letdown, primarily from the residual heat removal system (RHR) and to a lesser extent from the normal chemical and volume control system (CVCS) letdown. Coolant is returned to the primary system via the CVCS using one of the centrifugal charging pumps.

If the charging and the letdown flows are equal, RCS pressure remains constant. Any imbalance between charging and letdown causes the pressure in the RCS to either increase or decrease. As an example, increasing charging flow and maintaining the letdown flow constant, increases the mass of coolant in the RCS. Since water is virtually incompressible, and with the mass increasing pressure increases.

Operating in this configuration HCV-128 (the letdown isolation valve from the RHR system) is fully open. The control room operator manipulates PCV-131 in either automatic or manual to vary the amount of coolant letdown from the RCS. Additional letdown flow, from the CVCS, is available through the letdown orifices 8149 A, B, and C. Since the pressure in the RCS is low at this time letdown flow from this source is extremely low.

RCS overpressure protection is provided by the; CVCS letdown line relief valve located between the letdown orifice isolation valves and the containment letdown isolation valve (CV 8152), the RHR pump suction and discharge reliefs, and the pressurizer PORVs.

The pressurizer PORV Over Pressure Mitigation switches must be in the UNBLOCK position for these valves to actuate on a low pressure. These valves, PORVs, provide cold overpressure protection by discharging water from the pressurizer to the pressurizer relief tank (PRT). They actuate, open, if the pressure in the RCS exceeds the cold overpressure protection setpoint.

The nuclear steam supply system (NSSS) is in MODE 5, cold shutdown; T_{avg}<200°F, K_{eff}<0.99 and Rated Thermal Power is N/A. In addition the following conditions exist; RCS pressure is being maintained between 320 and 400 psig, the boron concentration sufficiently high to yield at least 1% ΔK/K shutdown margin, and the pressurizer is solid and reactor coolant pumps are secured.

Decay heat is removed from the core by the residual heat removal system with letdown established for RCS cleanup. The steam generators are in "wet layup" (filled to the 100% level with water) and all secondary systems are secured with the exception of one circulating water pump. The main turbine and both feedwater pump turbines are on their respective turning gears. All pre-startup checklists have been completed.

19.2.2 Operations

As required by the plants Technical Specifications, the reactor coolant must be at its no-load operating temperature prior to criticality. Increasing the temperature of the reactor coolant is accomplished by using decay heat from the reactor core and reactor coolant pump heat.

With the plant in cold shutdown, the pressure in the RCS must be at least 320 psig to operate the reactor coolant pumps. A pressure of 320 psig provides two functions; first it ensures a net positive suction head for the RCP and second it ensures that the minimum differential pressure required for lifting the number one seal is met. In addition, the pressure in the RCS must be maintained below 450 psig while the residual heat removal system is aligned to the reactor coolant system. This ensures that the low pressure piping of the RHR system is not over pressurized and prevents the lifting of the RHR pump discharge relief valves. Finally seal injection flow from the chemical and volume control system is required to cool and lubricate the lower radial bearing and the seal package of the RCP.

Prior to starting the reactor coolant pumps, a steam bubble is established in the pressurizer. This ensures that a surge volume is available in the pressurizer for any water expansion caused by the heatup of the coolant as it passes through the reactor coolant pump and mixes with the hotter water in the core.

Steam bubble formation, in the pressurizer, is accomplished by increasing the temperature of the coolant inside the pressurizer with the pressurizer heaters. Concurrently, charging and letdown flows are adjusted to maintain the pressure within the RCS between 320 and 400 psig.

When the temperature in the pressurizer reaches 428°F to 448°F (saturation temperature for an RCS pressure corresponding to 322 psig and 400 psig), charging flow is reduced to form the steam bubble. As the steam bubble forms, RCS letdown is increased and the charging flow is maintained constant. The difference between these flow rates causes the level in the pressurizer to decrease toward 25 percent. With a steam bubble established in the pressurizer, system pressure control is accomplished with pressurizer heaters and spray valve operation.

Nitrogen Bubble Pressure Control

RCS pressure control through use of a nitrogen cover gas can be used as an alternate method of controlling pressure. This control scheme may be used when the shutdown requires access to the RCS. In this mode of pressure control, as shown in Figure 19-2, the pressurizer is filled to 90 percent, as indicated by the cold calibrated level instrument, and it is vented to the pressurizer relief tank (PRT) via open PORV's.

A tygon hose is temporarily installed between the pressurizer vent line (upstream of the PORVs) and the vent on the PRT to supply nitrogen to the pressurizer. The nitrogen pressure regulator, located on the PRT, is adjusted so that six psig of nitrogen overpressure is applied to the PRT. This form of pressure control provides a stable RCS pressure without constant operator action, while still allowing some expansion and contraction of the coolant.

During nitrogen bubble pressure control, RHR flow is maintained for decay heat removal and, letdown and charging are maintained for purification of the reactor coolant. Nitrogen, is used as a cover gas to minimize corrosion and prevent explosive mixtures of hydrogen and oxygen within the RCS piping or components.

RCS overpressure-protection is provided by the letdown line relief (if aligned), the RHR pump suction and discharge reliefs, and rupture discs installed on the PRT. Since the RCS and the pressurizer are not completely filled with coolant, the chances of an overpressure event are less likely. The transformation of pressure control from nitrogen to steam is accomplished at the start of the plant heatup.

Prior to drawing a steam bubble, if charging and letdown is not in service then they must be placed in service. Letdown is via the RHR to CVCS cross-connect valve HCV-128. A letdown flow of 75 gallons per minute is established through combined use of HCV-128 (fully open) and throttling letdown pressure control valve PCV-131. When pressurizer level is < 80 percent, a charging pump is started to maintain pressurizer level constant during bubble formation.

All groups of pressurizer heaters are energized to raise pressurizer water temperature to saturation. Steam formation begins when pressurizer water temperature is at saturation. This corresponds to approximately 230°F at six psig. With the pressurizer at saturated conditions, further heating causes temperature and pressure in the pressurizer to increase. As pressure increases, steam and nitrogen are displaced to the PRT.

Indications of steam formation are provided by the pressurizer vapor space and the pressurizer PORV tail-pipe temperature instruments. Initially these instruments indicate containment ambient temperature. With steam formation they approach the saturation temperature of 230°F. Positive indication of a steam bubble is determined by the use of the auxiliary spray valve from the CVCS. Closing the PORVs and initiating auxiliary spray flow into the pressurizer, a rapid pressure decrease should be seen; if so, a steam bubble exists in the pressurizer. When this phenomenon is observed, the PORVs are placed in automatic and the connection between the PRT and pressurizer is isolated and the tygon tubing is removed. As pressure increases toward 320 psig, letdown flow is increased to accommodate the expansion of the reactor coolant and to lower the level in the pressurizer.

Once a bubble is drawn in the pressurizer and the pressure in the RCS is equal to 320 psig the reactor coolant pumps are started. The RCPs are started one at a time until all four pumps are running. After all reactor coolant pumps are operating, the residual heat removal pumps are turned off since they are no longer needed to remove decay heat or provide forced flow through the core as required the plants Technical Specifications.

With all four RCPs operating and the RHR system secured the reactor coolant begins to heatup at a rate of approximately 50°F per hour. The residual heat removal system alignment is maintained in its present configuration to provide adequate letdown to remove the excess coolant volume produced by expansion due to heatup. As a result of the heatup and draining of the pressurize, approximately one-third of the reactor coolant system volume (30,000 gallons) is diverted to the holdup tanks through the chemical and volume control system.

Prior to exceeding 200°F, entering MODE 4, all surveillance requirements for entering this MODE must be satisfactorily completed (See Checklist No.1). The plant's Technical Specifications define MODE 4 as Hot shutdown; 350°F> T_{avg} <200°F, K_{eff} <0.99 and Rated Thermal Power is N/A.

As the reactor coolant temperature approaches 200°F, steam generator draining is commenced through the normal blowdown system. If the oxygen concentration in the reactor coolant is high, hydrazine is added through the chemical and volume control system for oxygen scavenging (oxygen must be within specification before exceeding 250°F).

When the oxygen concentration is within specification, a hydrogen blanket (hydrogen overpressure) is established in the volume control tank. This is accomplished by securing the nitrogen regulator, opening the vent from the volume control tank to the waste gas header, and raising volume control tank level to force the nitrogen to the waste gas system. After the volume control tank level has risen to approximately 95%, the hydrogen regulator is placed in service and the last of the nitrogen is purged to the waste gas system. The volume control tank level is allowed to return to normal with the hydrogen regulator maintaining an overpressure of approximately 15 to 20 psig.

An inspection of containment is conducted for debris that could cause blockage of the containment sumps or produce fire hazards as the systems and their components heat up. Finally, before entering MODE 4, the containment spray system is aligned for operation. At approximately 220°F RCS temperature, steam formation begins in the steam generators. The nitrogen, used to prevent or minimize corrosion inside the steam generators, is isolated.

Secondary Plant Heatup

Warming of the steam lines and the main turbine is performed simultaneously with RCS heatup; however, before steam can be supplied to the main turbine, the turbine lubricating oil and cooling water systems must be placed in service. Prior to rolling the main turbine the turbine bearing oil system must be in service. Motor driven lubricating oil pumps provide the necessary lubrication for rolling the main turbine. To ensure even heating of the turbine rotor, the turbine must be on the turning gear prior to warming.

The condensate and circulating water systems are started at this time. The condensate and feedwater systems are aligned to provide cleanup of the secondary water by circulating condensate through one condensate polishing demineralizer. The condensate flows through the in-service polisher, through the low pressure heaters to the main feedwater pumps. The feedwater pumps are aligned for operation and the feed pump turbines are rotating on their respective turning gear. Condensate flows through the idle feedwater pumps, the high pressure heaters, and back to the low pressure condenser through the feedwater pump recirculation valve.

Warming of the steam lines is initiated by opening the main steam isolation valves (MSIV). This admits steam to the individual steam lines up to the main turbine stop valves. This allows the steam lines to be heated during RCS heatup. Condensate forming in the steam lines is drained to the condenser via steam line drain traps and drain trap bypasses. Simultaneous warming of the steam lines and the main turbine is accomplished by clearing all turbine generator trip signals and initiating turbine shell warming. The turbine hydraulic, gland sealing steam, and condenser air removal systems must be placed into service to provide steam valve control and condenser vacuum. Steam blanketing of the MSR's is also initiated to prewarm and deaerate the MSR first and second stage reheater tube bundles. MSR steam blanketing steam is supplied from the auxiliary steam system.

Before the reactor coolant temperature reaches 350°F, surveillance requirements for entering MODE 3, hot standby, must be satisfactorily completed (See Checklist No. 1). MODE 3 is defined by the plants Technical Specifications as; T_{avg} > 350°F, Keff <0.99, and Rated Thermal Power N/A.

The residual heat removal system is isolated from the reactor coolant system and aligned for at power operation (emergency core cooling system lineup). At this time, all reactor coolant letdown is through the normal letdown orifice pathway to the chemical and volume control system. After the residual heat removal system is isolated from the reactor coolant system, pressure is allowed to increase as the temperature in the pressurizer increases.

RCS Heatup and Pressurization

As plant heatup continues, pressurizer level continues to increase due to coolant expansion. Once placed in automatic, the pressurizer level control system automatically compensates by reducing charging flow. During this evolution the charging header valve, HCV-182, may require adjustment by the control room operator to maintain proper reactor coolant pump seal injection flow as the charging flow varies.

As reactor coolant pressure increases, letdown flow also increases. The low pressure letdown valve (PCV-131) is adjusted, closed, until the normal letdown pressure (350 psig) is achieved and the orifice isolation valves are closed as necessary to maintain letdown flow below the allowed maximum of 120 gpm. When reactor coolant system pressure reaches 1925 psig, the cold leg accumulator discharge valves are opened, and all emergency core cooling system equipment is checked for proper alignment.

Pressure and temperature relationships must be constantly observed during a combined plant heatup to prevent an inadvertent reactor trip or safety injection actuation signal. Although the reactor is shut down, a reactor trip signal would cause the shutdown rods to trip into the core. A safety injection signal would cause insertion of shutdown rods and activation of the design bases accident (DBA) sequencer. A safety injection can occur if RCS pressure is increased above 1915 psig (P-11 permissive). Above 1915 psig the block signal for low pressurizer pressure safety injection is automatically removed. Any evolution that would cause RCS pressure to decrease below 1807 psig initiates a low pressure safety injection.

It should also be understood that a safety injection will occur if the Hi Steamline flow setpoint is exceeded with S/G pressure below 600 psig or RCS T_{avg} below 553°F. The high steamline flow ESF signal could occur due to an increase in steamline flow, thru the steam dump valves or due HP turbine warming rate. Over-feeding a single steam generator, causes its pressure to decrease due to the cold feedwater addition. This could cause a High Steamline ΔP safety injection.

Primary plant heatup is terminated by automatic actuation of the steam dumps (in steam pressure control) when the pressure inside the steam generator reaches 1092 psig. RCS temperature remains constant at 557°F due to the steam dumps removing any additional heat generated in the primary system.

Turbine warming is secured after a specified warming time has been completed. Upon completion of heatup and pressurization, the reactor plant is at normal no load temperature and pressure ready for a reactor startup. The turbine and steam lines are warmed with preparations completed for a secondary startup. Pressurizer heaters and sprays are placed in automatic control when pressure reaches the normal operating value of 2235 psig.

A reactor coolant leak rate test is performed when the reactor coolant temperature is greater than 400°F and pressure is equal to 2235 psig. This test is required at this time only if the reactor coolant system has been opened for refueling or maintenance.

Plant conditions are as follows:

- The RCS is at normal operating temperature and pressure (557°F, and 2235 psig)
- The reactor is shutdown
- Normal condenser vacuum
- Steam dump to the condenser in the steam pressure mode
- Main and feedwater pump turbines on the turning gear and,
- All electrical power is supplied from off-site.

19.3 Reactor Startup To Minimum Load

A reactor startup is performed at a no-load temperature of 557°F. The Technical Specifications require that a minimum temperature for criticality must be met. For most plants this temperature is 551°F. If the average temperature of the reactor coolant drops below this value, the operator has an allotted time frame in which the temperature must be restored above the minimum temperature for criticality or the rods must be fully inserted into the core.

Prior to the withdrawal of the control rods to criticality, an estimated critical condition (ECC) is calculated. The estimated critical condition calculation is a reactivity balance comparing previous critical conditions and reactivity changes following a shutdown to the desired conditions for the next startup. Since the previous and present reactivity conditions are known, it is relatively easy to calculate the required conditions for the next criticality. The reactivity variables for which the operator has direct control are boron concentration and control rod position. The ECC is normally computed using a desired critical rod position. Boron concentration is then adjusted so that the reactor will be critical at the desired rod position.

Prior to entering MODE 2, startup, and MODE 1, power operation, all surveillance requirements for changing modes must be satisfactorily completed (See Checklist No. 2). MODE 2 and MODE 1 are defined by the Technical Specifications as MODE 2 Keff $\geq$ .99 Percent Rated Thermal Power $\leq$ 5%, and MODE 1 Keff $\geq$ .99 Percent Rated Thermal Power $\geq$ 5%.

Control rods are withdrawn to establish criticality. Its difficult to identify exactly when criticality is achieved during a startup; however, it is readily apparent when supercritical conditions are reached. Supercriticality is indicated by a constant positive startup rate and steadily increasing count rate with no control rod withdrawal. Criticality is declared when these conditions are observed. After criticality is achieved, control rods are positioned to increase nuclear power to 10^{-8} amps in the intermediate range. When power exceeds the "Source Range Block Permissive" (P-6) setpoint, the source range trip is manually blocked and the source range high voltage supply is deenergized. Power is stabilized at 10^{-8} amps to record actual critical conditions. Data recorded consists of control rod positions, boron concentration, and RCS loop average temperatures.

After recording the critical data, control rods are positioned to increase power to the point of adding heat, approximately one percent power as indicated on the power range instruments, and then stabilizing power at two percent. As nuclear power increases above the point of adding heat, fuel and moderator temperatures will increase. This results in negative reactivity addition due to doppler and moderator temperature coefficients.

As the moderator temperature increases, the coolant expands into the pressurizer causing increased level and pressure. Also, the steam generator temperature and pressure begin to increase. Since the steam dumps are in steam pressure mode of control, the steam dump valves modulate open maintaining steam generator pressure at 1092 psig. The steam pressure of 1092 psig is the pressure for a saturation temperature of 557°F which is the no-load T_{avg}. With the steam dumps in steam pressure mode, RCS T_{avg} will not follow programmed T_{avg}, but will increase to produce the ΔT required from primary to secondary to support two percent of full power steam flow.

As the steam dumps modulate open, increasing steam flow to the main condenser, feedwater flow to the steam generators must be increased. Since the motor driven auxiliary feedwater pump (non-safety grade) can supply only about two percent of rated feed flow, care must be exercised to ensure power level does not exceed two percent.

At some point between 1 x 10^{-8} amps and two percent power, anticipated reactivity changes for power escalation, due to power defect and xenon, are determined, and the required boron concentration change is started. The rate of power increase could be limited by improper control rod position if power defect and xenon changes are not predicted correctly.

Power level is maintained at two percent while a main feedwater pump is started and aligned to supply the steam generators through the main feed regulating bypass valves. The main feedwater bypass valves (manually operated) control steam generator level. Main feed pump speed control is maintained in manual with 150-300 psid between steam and feed pressures. This high ΔP is required to ensure adequate feed flow to the S/Gs through the bypass valves. With a main feedwater pump feeding the steam generators and the auxiliary feedwater system in its safeguards lineup, turbine startup may proceed.

To minimize primary plant transients, the turbine is rolled with reactor power between 10 and 15 percent power. This is accomplished by withdrawing the control rods to raise power and RCS temperature. Steam generator temperature and pressure will attempt to increase with RCS temperature; however, the steam dumps modulate open to maintain S/G pressure constant and creating the steam demand which changes the ΔT across the core.

With nuclear power at 15 percent; the main turbine is rolled off the turning gear and up to 1800 rpm. As steam flow to the turbine increases, the steam dumps modulate close to maintain 1092 psig steam generator pressure. At 1800 rpm, the turbine auxiliary systems are aligned for normal operation, and the turbine protection system is tested to verify proper operation. Also at this time, the moisture seperator reheater (MSR) steam control is placed into automatic. This removes auxiliary steam supply from MSR steam blanketing and allows operation of the first and second stage MSR steam supplies. Generator synchronization is performed after satisfactory turbine protection system testing.

When the main generator voltage, frequency and phase are matched with that of the grid, the generator output breakers can be closed, and the generator picks up a minimum load of approximately 60 MWE. When the generator breakers close the turbine EHC automatically shifts from speed control to load control and the unit is now ready to increase the electrical output to the grid.

19.4 Power Operations

The turbine loading rate is determined by the most limiting of the following; turbine loading limit curve, fuel pre-conditioning requirements, or the load dispatchers request. Once the loading rate is determined the turbine load is increased to 180 MWe. This load corresponds to approximately 15 percent turbine load. Increasing turbine load to 180 MWe results in a shift of steam flow from the steam dumps to the turbine with little or no change in nuclear power. When the steam dumps are fully closed they are placed in the T_{avg} mode of operation.

Rod control is placed into automatic when turbine load is sufficient to satisfy C-5 (15 percent turbine load) and T_{avg} is within 1 °F of T_{ref}. Since the main feedwater bypass valves are rated for 20-25 percent of full feedwater flow, the main feedwater regulating valves should be placed in service prior to increasing load above 20 percent. Main feed pump speed control can be placed in automatic when the main feed regulating valves are controlling S/G levels.

To prevent axial Xenon oscillations and to maintain peaking factors within license limits, the operator must insure that the indicated axial flux difference is maintained within the limits illustrated in Figure 19-3. Some nuclear units may operate with a target flux band which is more restrictive than the limits shown in this figure. Operating outside of a flux band results in the accumulation of "penalty minutes" as defined in Technical Specifications. If 60 or more penalty minutes are accumulated in the previous 24 hours, operation above 50% thermal power is prohibited.

Penalty minutes are accumulated at the rate of one penalty minute for each minute outside the band while above 50% power, and one half a penalty minute for each minute outside the band while below 50% power. These restrictions have resulted in major changes in plant operations.

1. To maintain power distribution within the limits of Figure 19-3, the control rods must be maintained nearly fully withdrawn during all phases of power operations (except for short-term transients). During a power escalation,

since the rods must be fully withdrawn, the power defect associated with the increase must be overcome by boron dilution.

2. The dilution of boron out of the reactor coolant is a slow process. Therefore, power changes at sustained rates of greater than 1% or 2%/minute are not possible if the axial flux distribution is to be maintained within the band.

During power escalations from 20 percent to 100 percent power, additional secondary pumps must be placed into service. These include condensate pumps, circulating water pumps, service water pumps, feedwater pump, and heater drain tank pumps. Condensate demineralizer polishers are also sequentially placed into service as power increases.

The load on the turbine is increased by selecting the desired loading rate and pressing the load increase push-button to increase load set to the desired value. The main turbine control valves modulate to a position which should produce the megawatt output selected with the load set controller. A first stage pressure feedback signal adjusts the control valve signal slightly for a linear load increase as the control valves open.

The Single Loop Loss of Flow permissive (P-8) enables the single loop loss of flow reactor trip when reactor power exceeds 39%.

At approximately 50% load, a calorimetric (heat balance) is performed on the secondary system, and an adjustment to the power range nuclear instruments is performed if necessary. Further calorimetrics are performed at 90% and 100% power. These additional calorimetrics are performed to ensure that the power range nuclear instruments are properly calibrated.

At all times when the reactor is critical, the control banks must be maintained withdrawn above their respective insertion limits. All shutdown banks and control banks A and B must be fully withdrawn while control banks C and D must be withdrawn to a position that is greater than the limit specified in Figure 19-4. Maintaining the rods above their respective insertion limits ensures that sufficient negative reactivity is available, from the rods, to achieve and maintain the required shutdown margin in the event of a reactor trip.

19.5 Plant Shutdown

Plant shutdown is accomplished by essentially reversing the steps described in plant startup.

APPENDIX 19-1
PLANT STARTUP FROM COLD SHUTDOWN

I. INITIAL CONDITIONS

1. Cold Shutdown - MODE 5
    - $K_{eff} < 0.99$
    - 0% Rater Thermal Power
    - T_{avg} <200 °F
2. Reactor Coolant System solid.
3. RCS Temperature 150 - 160°F.

Note:
Temperature may be less than 150 °F depending upon decay heat load of the core.

4. RCS Pressure 320 - 400 psig.
5. Steam Generators Filled to Wet-Layup.
    - 100% Level Indication
6. Secondary Systems Shutdown.
    - Main Turbine and Feedwater Pump Turbines on their Turning Gears
7. Pre-Startup Checklists Completed.

II. INSTRUCTIONS

A. Heatup from COLD SHUTDOWN to HOT SHUTDOWN (MODE 5 to MODE 4).

1. Permission received from Operations Supervisor for startup.
2. Begin establishing steam generator water levels to 33 ± 5% on narrow range indication.
3. Verify or establish RCP seal injection flow.

CAUTION:
Do not exceed a heatup rate of 100 °F/hr on the pressurizer, 100 °F on the RCS or 320 °F ΔT between pressurizer and spray temperature. Use auxiliary sprays for pressurizer and RCS mixing.

4. Energize pressurizer heaters and begin pressurizer heatup.
5. Establish a pressurizer steam bubble by;
    a. Increasing pressurizer temperature using pressurizer heaters.
    b. Adjust charging and letdown flow to maintain pressurizer pressure at approximately 320-400 psig while reducing pressurizer level.
    c. As pressurizer temperature approaches 428°F (saturation temperature for 320 psig), reduce pressurizer level toward 25%.
6. Start the reactor coolant pumps. After running for five minutes, sample the RCS for chemistry specifications. Partially open pressurizer spray valves for mixing.
7. Maintain the RCS temperature <160°F by adjusting flow through the RHR heat exchangers.

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PLANT STARTUP FROM COLD SHUTDOWN

> **NOTE:**
> *The 160 °F limit is based on cold water addition accident. This limits ΔT between seal injection water accumulating in the intermediate leg and the remainder of the RCS.*

8. Stop RHR pumps.
9. Allow RCS temperature to increase to 200°F.
10. When RCS temperature reaches 200°F, ascertain that primary system water chemistry is within specifications.
11. When pressurizer level is at the no-load operating level (25%), place the pressurizer level control system in automatic.
12. Verify shutdown rods are withdrawn and that sufficient SHUTDOWN MARGIN is available.
13. Pre-startup checklist for license requirements are completed. (See Checklist 1)
14. At 200°F RCS temperature, establish a hydrogen blanket in the volume control tank.
15. Open main steam isolation valves and warm main steam lines.
16. Continue pressurizer heatup to maintain desired pressure. Use low pressure letdown control valve to maintain letdown flow. Once a steam bubble is established in the pressurizer, RCS pressure will be controlled be heater and spray actuation.
17. Condensate cleanup is in progress.
18. When condensate chemistry is within specifications, align the condensate and feedwater systems to a normal at-power configuration.
19. Verify condenser vacuum.
20. Warm the main turbine.

> **CAUTION:**
> *Prior to reaching 350 °F in the RCS, perform or verify the following:*
> - *Verify the control rod drive cooling fans are operating*
> - *Terminate residual heat removal letdown to the CVCS*

B. Heatup from HOT SHUTDOWN to HOT STANDBY (MODE 4 to MODE 3).

1. Complete startup checklist for license requirements. (See Checklist No.1)
2. Complete the ECCS Master Checklist (SI, RHR, etc. aligned).
3. Maintain RCS pressure is within the operating band of the Pressure-Temperature Limitations Curve.

> **NOTE:**
> *As the RCS pressure increases, maintain letdown flow at a maximum 120 gpm by increasing the setting of the back pressure regulator (PCV-131) until pressure reaches ~350 psig and then by closing the letdown orifice isolation valves as necessary.*

4. Prior to reaching 1000 psig, open each cold leg accumulator isolation valves and then de-energize power to each valve.
5. When RCS pressure exceeds 1915 psig, verify pressurizer low pressure safety injection logic automatic reset.

6. When RCS temperature reaches 553°F, verify high steamline flow safety injection logic automatic reset.
7. Maintain RCS temperature by use of the steam dump system in the pressure control mode.
8. Establish HOT STANDBY conditions of 557°F T_{avg}.

C. Heatup from HOT STANDBY to POWER OPERATIONS (MODE 3 to MODE 1).

NOTE:
See Checklist Number 2.

1. Administrative permission to take the reactor critical has been obtained.
2. Notify system dispatcher of unit startup and approximate time when the generator will be tied to the grid.
3. Notify onsite personnel of the reactor startup over the public address system.
4. Perform reactor trip breaker check (if not performed during the previous seven days).
5. Calculate the estimated critical boron concentration for the desired critical control bank rod position.
6. If necessary, conduct a boron concentration change to the estimated critical boron concentration. Equalize boron concentration between the reactor coolant loops and the pressurizer by turning on the pressurizer backup heaters.

NOTE:
Nuclear instrumentation shall be monitored very closely in anticipation of unplanned reactivity rate of change.

NOTE:
Block the source range high flux level at shutdown alarm at both source range panels.

7. Verify all shutdown rods are fully withdrawn within 15 minutes of withdrawing control rods.
8. If the shutdown banks are inserted, complete a shutdown margin calculation (assuming shutdown banks are out) to preclude an inadvertent MODE change during rod withdrawal. Withdraw the shutdown banks to the full out position.

NOTE:
Nuclear instrumentation shall be monitored very closely in anticipation of unplanned reactivity rate of change.

9. Verify proper operation of all reactor coolant pumps.
10. Withdraw the control bank rods in manual and take the reactor critical.

APPENDIX 19-1
PLANT STARTUP FROM COLD SHUTDOWN

11. If the control bank height at criticality is below the minimum insertion limits for the 0 percent power conditions.
    a. Reinsert all control bank rods to the bottom of the core.
    b. Recalculate the ECC.
    c. Borate to the new ECC concentration.
    d. Withdraw the control bank rods in manual and take the reactor critical.
12. Withdraw rods to increase reactor power. Block the source range trip at P-6 (10^{-10} amps on the intermediate range). Take critical data at $1x10^{-8}$ amps (typically).
13. Withdraw rods to bring reactor power to approximately 2% on the power range indicators and select the highest power range channel to be recorded on the NR 45 recorder.
14. Check that the steam dump pressure controller is set at 1092 psig and turned to the ON position.
15. If not running, start a main feedwater pump at 2% power. Maintain steam generator levels at 33% ± 5% narrow range level indication during secondary plant startup by throttling the feedwater bypass regulating valves.

CAUTION:
Coordinate all steam generator steam removal and significant feedwater changes with the Reactor Operator while rod control is in manual.

16. Verify the turbine has been on the turning gear for at least one hour.
17. Increase reactor power by manual adjustment of the control bank until the steam dump is bypassing steam flow equivalent to 10-15 percent nuclear power.
18. When reactor power increases above 10 percent, verify the "Nuclear At Power" Permissive P-10 light comes on and the "At Power" Permissive P-7 light goes out.

NOTE:
The "At Power" reactor trips are now enabled.

19. Manually block the intermediate range reactor trip and the power range low setpoint reactor trip after P-10 has been actuated.
20. Verify the main and unit auxiliary transformer cooling systems are aligned for automatic operation.
21. Roll the main turbine to 1800 rpm, synchronize the generator and connect it to the grid.
22. Increase generator load at the desired rate while maintaining T_{avg} by manual rod control.

CAUTION:
Maintain programmed steam generator level during the following step.

23. At approximately 15% power transfer feedwater flow control from the bypass valves to the main regulating valves.
24. When turbine power has increased above (~180 MWe) 15% (C-5), transfer the rod control system to automatic.

APPENDIX 19-1
PLANT STARTUP FROM COLD SHUTDOWN

25. After the rod control is placed in automatic, check steam pressure less than steam dump set point (1092 psig), all steam dump valves closed, then transfer steam dumps to T_{avg} control mode.
26. Transfer steam generator feedwater regulating control valves and feed pump speed control to automatic when above 15% power.
27. Place additional secondary equipment in service to support load increase.
28. At 50 percent power, place remainder of secondary pumps in service.
29. Perform secondary calorimetric (heat balance) for nuclear instrumentation adjustment as required.

Checklist Number 1

> **NOTE:**
> *This listing is typical of the TECHNICAL SPECIFICATIONS requirements for changing MODES during a plant heatup. This is not a comprehensive list.*

REACTIVITY CONTROL (Section 1) Limiting Condition for Operations:

Shutdown Margin for $T_{avg} > 200°F$
Charging Pumps
Rod Position Indication Systems
Boration Systems Flowpaths
Borated Water Sources

INSTRUMENTATION (Section 3) Limiting Condition for Operations:

Reactor Trip Instrumentation
Engineered Safety Features Actuation System Instrumentation
Other Required Instruments

REACTOR COOLANT SYSTEM (Section 4) Limiting Condition for Operations:

Reactor Coolant System
Pressurizer Safety Valves, Relief Valves
Steam Generators
RCS Pressure/Temperature Limits
Pressurizer
Pressurizer Heaters
RCS Leakage, Chemistry, Activity
RCS Structural Integrity

EMERGENCY CORE COOLING SYSTEMS (Section 5) Limiting Condition for Operations:

ECCS Subsystems
Accumulators (RCS pressure is > 1000 psig)
Refueling Water Storage Tank
Boron Injection System

CONTAINMENT SYSTEMS (Section 6) Limiting Condition for Operations:

Integrity
Internal Pressure
Containment Spray
Containment Isolation Valves
Leakage
Air Temperature
Spray Additive System
Air Locks
Ventilation
Containment Cooling

PLANT SYSTEMS (Section 7) Limiting Condition for Operations:

Steam Generator Safety Valves
Component Cooling Water
Condensate Storage Tank
Auxiliary Feedwater System
Essential Service Water
Main Steamline Isolation Valves

ELECTRICAL POWER SYSTEMS (Section 8)Limiting Condition for Operations:
AC Sources
Vital AC & DC Instrumentation Power
AC Distribution System

Checklist Number 2

NOTE:
This listing is typical of the TECHNICAL SPECIFICATIONS requirements for changing MODES during reactor startup and power operations. This is not a comprehensive list.

REACTIVITY CONTROL (Section 1) Limiting Conditions for Operation:

Moderator Temperature Coefficient
Minimum Temperature for Criticality
Movable Control Assemblies
Rod Drop Time
Shutdown and Control Banks - Rod Insertion Limits

POWER DISTRIBUTION LIMITS (Section 2) Limiting Conditions for Operation:

Axial Flux Difference
Heat Flux Hot Channel Factor Fq(Z)
RCS Flowrate
Quadrant Power Tilt Ratio
DNB Parameters

INSTRUMENTATION (Section 3) Limiting Conditions for Operation:

Reactor Trip Instrumentation
Engineered Safety Features Actuation System
Movable Incore Detectors

REACTOR COOLANT SYSTEM (Section 4) Limiting Conditions for Operation:

RCS Loops and Coolant Circulation

CONTAINMENT SYSTEMS (Section 6) Limiting Conditions for Operation:

Hydrogen Analyzers
Hydrogen Recombiners
Containment Purge System

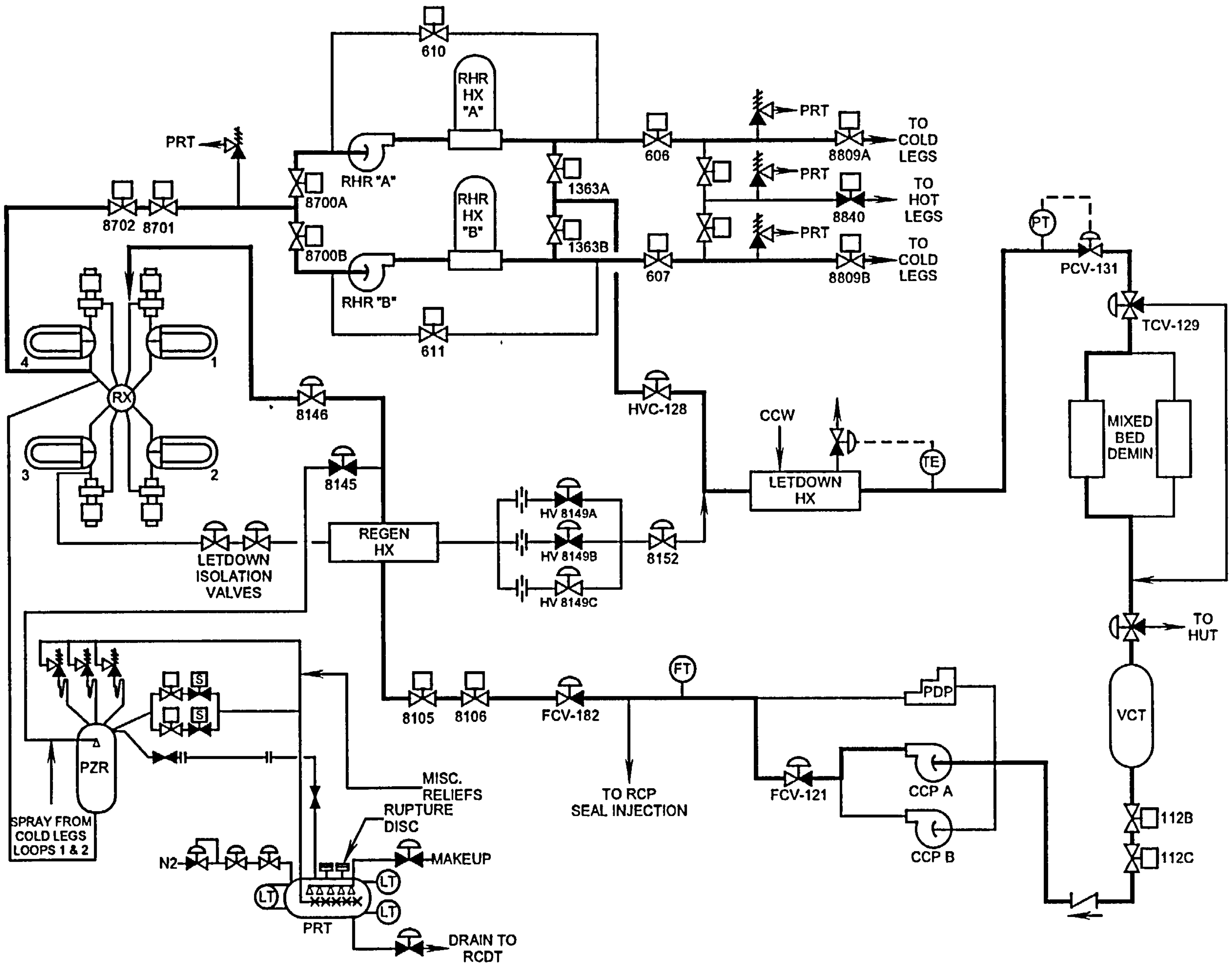

Figure 19-1 Solid Plant Pressure Control

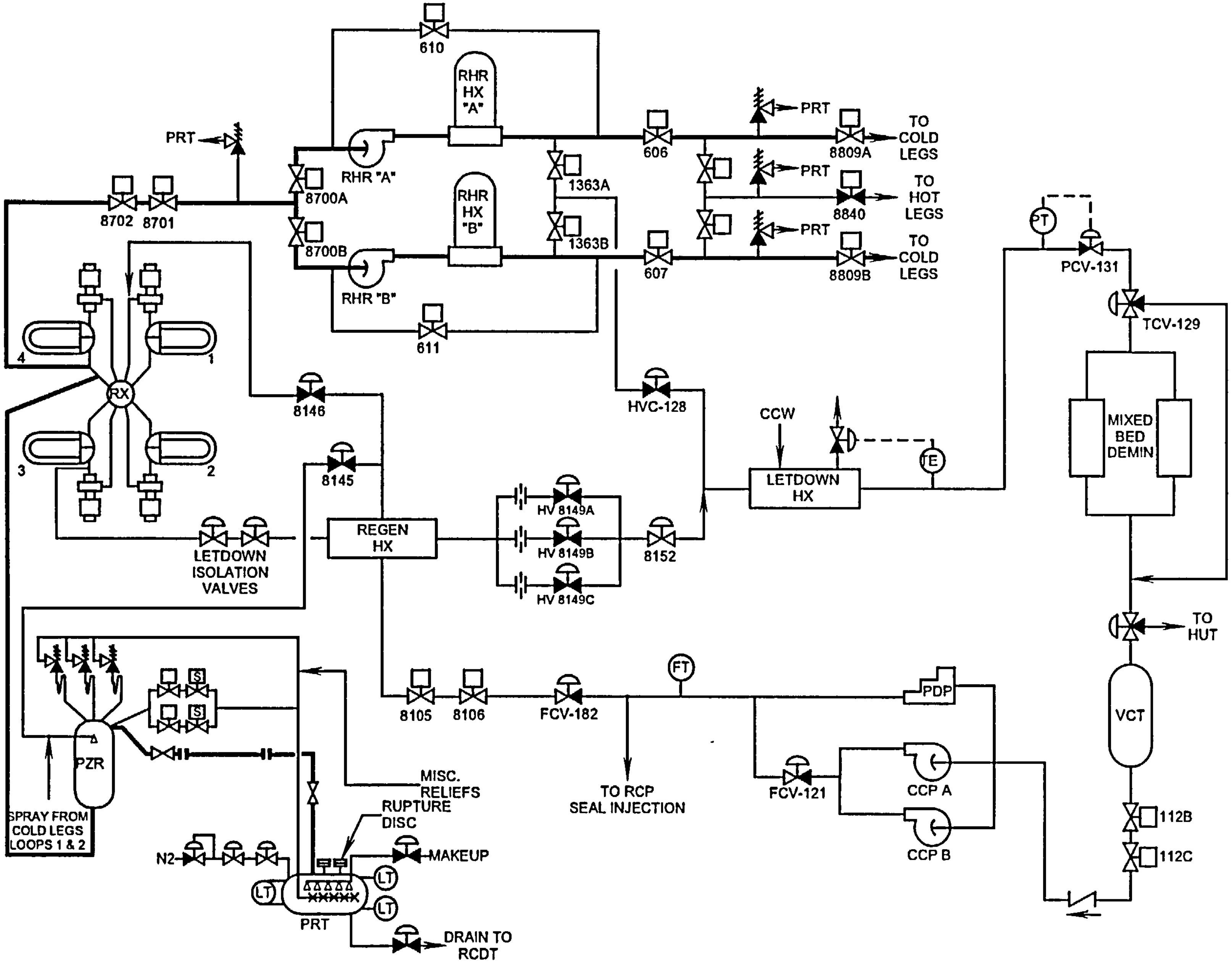

Figure 19-2 Nitrogen Bubble Pressure Control

0199-3
19-1

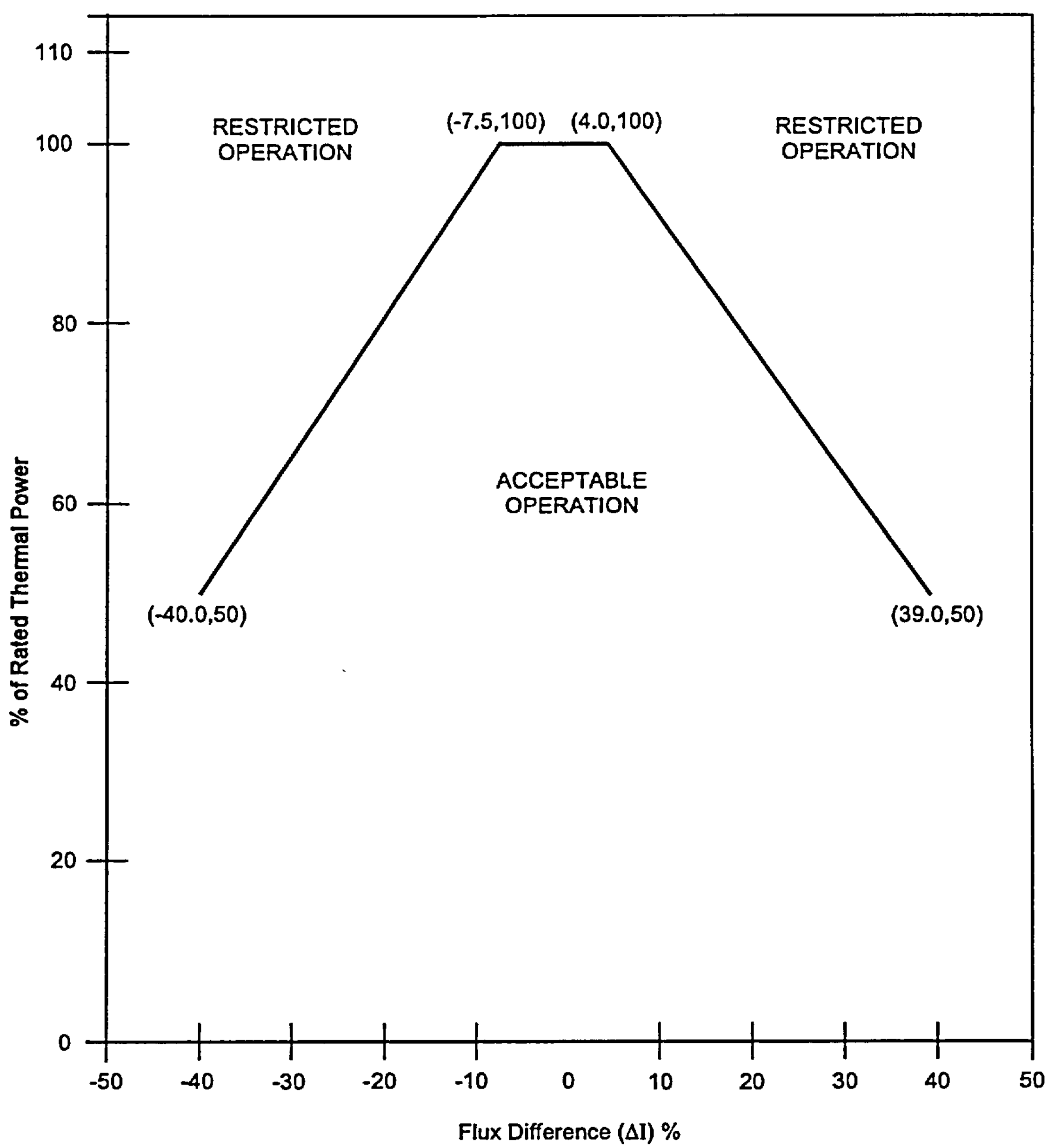

Figure 19-3 Axial Flux Difference Limits

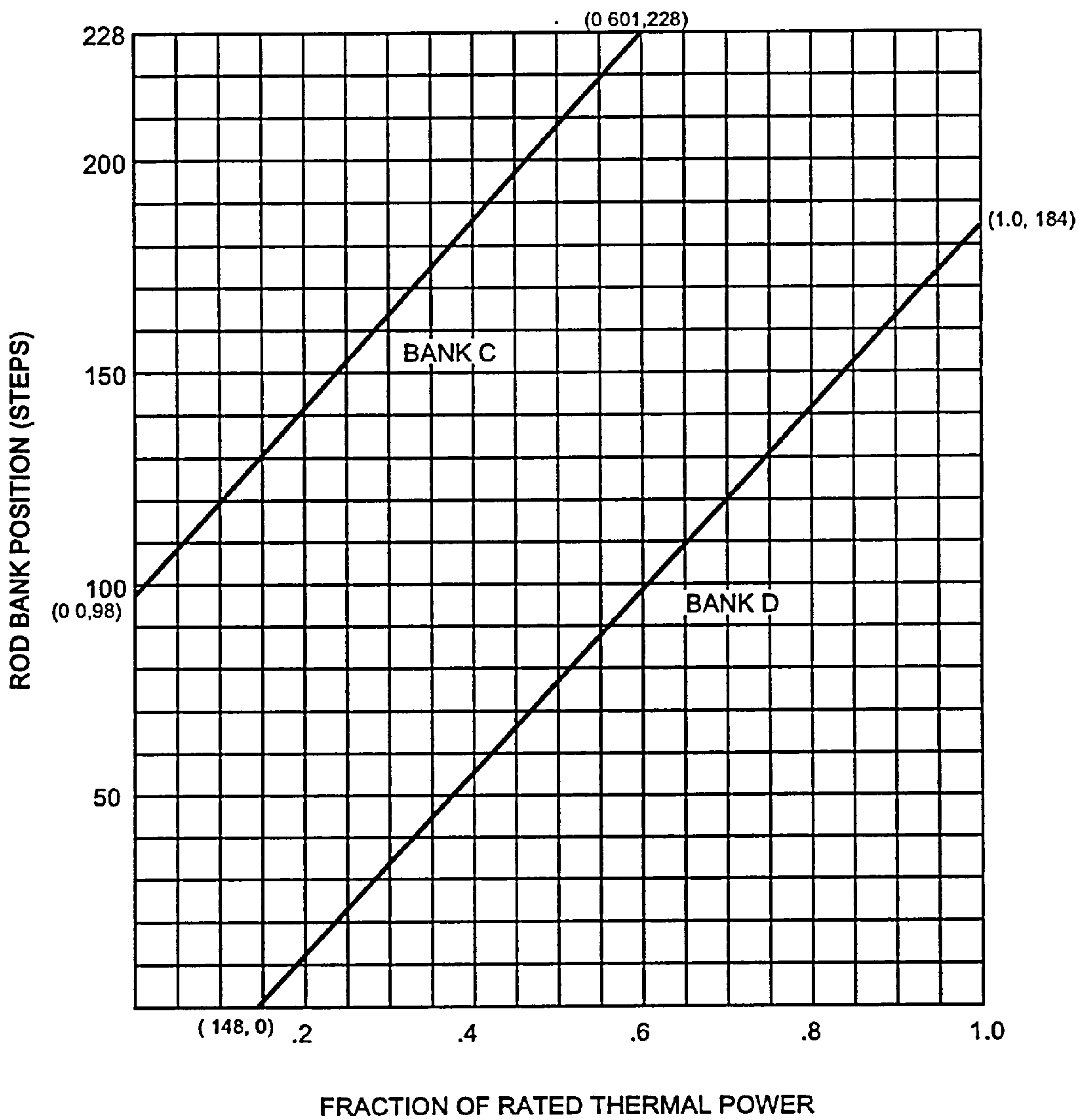

Figure 19-4 Rod Insertion Limits vs. Thermal Power

Westinghouse Technology Systems Manual

Appendix A

Learning Objectives

1.2 INTRODUCTION TO PRESSURIZED WATER REACTOR GENERATING SYSTEMS

Learning Objectives:

1. Define the following terms:

    a. Primary cycle,
    b. Secondary cycle, and
    c. Reactor coolant ststem (RCS) average temperature (T_{avg}).

2. Explain why T_{avg} is programmed to increase with an increasing plant load.

2.1 REACTOR PHYSICS REVIEW

Learning Objectives:

1. Define the following terms:

   a. K_{eff}
   b. Reactivity
   c. Reactivity coefficient
   d. Power defect
   e. Poison
   f. Critical
   g. Supercritical
   h. Subcritical
   i. Startup rate

2. Describe the following reactivity coefficients and explain how their values change with core life and reactor power level:

   a. Moderator temperature
   b. Fuel temperature (Doppler)
   c. Void
   d. Power

3. Explain the relative effects of the following poisons in plant operations:

   a. Xenon
   b. Samarium

4. Explain how the following controllable poisons affect core reactivity:

   a. Control rods
   b. Chemical shim

3.1 REACTOR VESSEL AND INTERNALS

Learning Objectives:

1. State the purpose of the following major reactor vessel and core components:

   a. Internals support ledge
   b. Neutron shield pad assembly
   c. Secondary support assembly
   d. Internals packages
   e. Neutron sources
   f. Burnable poison rod assemblies
   g. Thimble plug assemblies
   h. Irradiation specimens

2. Describe the flow path of reactor coolant from the inlet nozzles to the outlet nozzles of the reactor vessel.

3. List the core bypass flow paths.

4. Describe the physical arrangement of the following assemblies including the component parts listed:

   a. Fuel assembly
      - fuel rods
      - spring clip grid assembly
      - guide thimbles
      - top and bottom nozzles
   b. Control rod assembly
      - rodlets
      - spider
      - hub
      - drive shaft
   c. Rod drive mechanism
      - magnetic coils
      - gripper latches
      - pressure boundary

5. Describe the reactor vessel head seal arrangement.

6. Describe how the reactor vessel is supported.

3.2 REACTOR COOLANT SYSTEM

Learning Objectives:

1. State the purpose of the Reactor Coolant System (RCS).

2. List and state the purpose of the following RCS penetrations:

   a. Hot Leg (T_h)
      1. Pressurizer surge line
      2. Residual Heat Removal (RHR) suction
      3. Sample line
      4. RHR recirculation/Safety Injection (SI)

   b. Intermediate Leg
      1. Elbow flow taps
      2. Chemical and Volume Control System (CVCS) letdown
      3. Loop drain

   c. Cold Leg (T_c)
      1. Pressurizer spray line
      2. CVCS charging
      3. Common injection penetration for RHR, SI, and Cold Leg Accumulator
      4. High head injection
      5. Excess letdown

3. Describe the primary and secondary flow paths in the steam generator.

4. State the purpose of the following components of the reactor coolant pump.

   a. Thermal barrier heat exchanger
   b. Seal package
   c. Flywheel
   d. Anti-reverse rotation device
   e. Number 1 seal bypass valve
   f. Number 1 seal leak off valve
   g. Seal stand-pipe

5. Explain why seal injection flow is supplied to the reactor coolant pumps.

6. State the purpose of the following:

   a. Pressurizer
   b. Code safety valves
   c. Power operated relief valves (PORV)
   d. PORV block valves
   e. Pressurizer relief tank (PRT)
   f. Pressurizer spray valves
   g. Pressurizer heaters

7. Describe the methods for determining pressurizer relief and safety valve position and/or leakage.

8. Explain the following:

   a. Pressurizer spray driving force
   b. Purpose of pressurizer spray bypass

9. Explain how failure of the following components could lead to core damage.

   a. Reactor coolant pump seals
   b. Power operated relief valves

4.1 CHEMICAL AND VOLUME CONTROL SYSTEM

Learning Objectives:

1. List the purposes of the Chemical and Volume Control System (CVCS).

2. List in flow path order and state the purpose of the following major components of the CVCS:

   a. Regenerative heat exchanger
   b. Letdown orifice
   c. Letdown heat exchanger
   d. Demineralizers (ion exchangers)
   e. Letdown filter
   f. Volume control tank (VCT)
   g. Charging pump

3. Identify the components in the CVCS that are used to purify the reactor coolant and the types of contaminants each is designed to remove.

4. Describe how the makeup system is used to borate, dilute, and makeup a blended flow of boric acid to the reactor coolant system (RCS).

5. Explain why and for what plant conditions the following chemicals are added to the RCS:

   a. Lithium hydroxide
   b. Hydrogen
   c. Hydrazine

6. Describe the emergency boration flow path, and identify the plant conditions which would require its use.

7. State the purpose of the connection between the Residual Heat Removal System (RHR) and CVCS letdown.

8. List the plant operations that result in large amounts of influent into the Boron Recovery System (BRS).

9. Identify the changes in the CVCS that occur upon the receipt of an Engineered Safety Features Actuation Signal (ESFAS).

10. Explain how the CVCS is designed to prevent the following:

    a. Flashing and pressure transients in the regenerative and non-regenerative heat exchangers.
    b. Loss of suction to the charging pumps.
    c. High temperature in the letdown ion exchangers (demineralizers).
    d. Over and under pressurization of the volume control tank.

11. State when and why excess letdown would be used.

12. List the automatic actions initiated by VCT level instrumentation.

5.0 INTRODUCTION TO ENGINEERED SAFETY FEATURES

Learning Objectives:

1. State the purposes of the Engineered Safety Feature (ESF) Systems.

2. Identify the three barriers designed to limit the escape of fission products to the environment.

3. Explain the following terms:
   a. Redundancy
   b. Physical and electrical separation
   c. Regulatory terms associated with safety
      - Important to Safety
      - Safety Related
      - Safety Grade
   d. Seismic Category I
   e. Diversity
   f. Single Failure
   g. Active Failure
   h. Passive Failure
   i. ESF Train

4. List the four design categories or conditions of operation, and give an example of each.

5. List the five acceptance criteria for the Emergency Core Cooling Systems (ECCS).

5.1 RESIDUAL HEAT REMOVAL SYSTEM

Learning Objectives:

1. State the purposes of the Residual Heat Removal (RHR) System.

2. Describe the RHR system flow path including suction supplies, discharge points and major components during decay heat removal.

3. Describe the normal, at-power line-up of the RHR system.

4. Explain why Reactor Coolant System (RCS) pressure and temperature limits are placed on the initiation of RHR cooldown.

5. Explain how the RHR system is protected against over pressurization.

6. Explain how an intersystem LOCA is initiated in the residual heat removal system and what affect it can have on long-term core cooling.

5.2 EMERGENCY CORE COOLING SYSTEMS

Learning Objectives:

1. Explain why Emergency Core Cooling Systems (ECCS) are incorporated into plant design.

2. State the purpose of the following systems:
   a. Accumulator injection system,
   b. Safety Injection (SI) pump system,
   c. High head injection system, and
   d. Residual Heat Removal (RHR) system

3. State the purpose of the following major components:
   a. Refueling Water Storage Tank (RWST), and
   b. Containment recirculation sump

4. List the order of ECCS injection during the following abnormal conditions:
   a. Small (slow depressurization) loss of coolant accident, and
   b. Large loss of coolant accident.

5.3 CONTAINMENT

Learning Objectives:

1. State the purposes of the containment.

2. Briefly describe the function of the following:

   a. Containment liner
   b. Primary shield wall
   c. Secondary shield wall
   d. Refueling canal
   e. Containment sumps
   f. Containment recirculation sump
   g. Containment hydrogen analyzer

3. Briefly describe the methods of monitoring the containment environmental conditions.

5.4 CONTAINMENT TEMPERATURE PRESSURE AND COMBUSTIBLE GAS CONTROL

Learning Objectives:

1. State the purposes of Containment Ventilation Systems.
2. List the signals that automatically initiate purge and exhaust system isolation.
3. State the purposes of the Containment Spray System.
4. List the signals that automatically initiate the Containment Spray System.
5. State the purpose of the containment Hydrogen Recombiners.

5.6 CONTAINMENT PENETRATION AND ISOLATION SYSTEMS

Learning Objectives:

1. State the purposes of the system.

2. Define the following:

    a. Containment Integrity
    b. Containment Isolation Phase A
    c. Containment Isolation Phase B

3. List the signals that initiate phase A and phase B isolation.

5.7 AUXILIARY FEEDWATER SYSTEM

Learning Objectives:

1. State the purposes of the Auxiliary Feedwater System (AFW).

2. List all suction sources for the AFW pumps and under what conditions each is used.

3. List the five plant conditions that will result in an automatic start of the AFW system.

4. Explain how decay heat is removed following a plant trip and loss of offsite power.

5.. Explain the relationship between the auxiliary feedwater system and the pressurizer Power Operated Relief Valves (PORVs) during a loss of offsite power.

6.0 ELECTRICAL DISTRIBUTION SYSTEM

Learning Objectives:

1. State the purposes of the electrical distribution system.

2. Define the following:

    a. Offsite distribution
    b. Onsite distribution
    c. Class 1E distribution
    d. Standby power source
    e. Preferred power source.

3. Explain how a reliable source of power is ensured to the following:

    a. The 4.16-kv and 480-vac safety related distribution buses
    b. The 125-vdc instrumentation and control buses
    c. The 120-vac instrumentation buses.

4. Describe the response of the non-Class 1E electrical distribution system to a turbine-generator trip.

5. Describe the operation of the Class 1E electrical distribution system for all combinations of loss of offsite power and engineered safety features actuation signal.

6. List the automatic start signals for the emergency diesel generators.

7. Explain the effect of an automatic start signal on the diesel generator protection system.

8. Explain why a station blackout in conjunction with a reactor coolant pump seal failure is a high contributor to core damage frequency.

7.1 MAIN AND AUXILIARY STEAM SYSTEMS

Learning Objectives:

1. State the purposes of the main and auxiliary steam systems.

2. Identify the portion of the main steam system that is Seismic Category I.

3. List the components and connections located in the Seismic Category I portion of the main steam system and explain the purpose of each.

4. Explain how the main steam system design prevents the blowdown of more than one steam generator during a steam line rupture accident.

5. Describe the steam flowpaths for decay heat removal after a plant trip:

   a. With offsite power available
   b. Without offsite power available.

7.2 CONDENSATE AND FEEDWATER SYSTEM

Learning Objectives:

1. List in proper flowpath order and state the purpose of the following condensate and feedwater system components:

   a. Condenser
   b. Condensate (hotwell) pumps
   c. Demineralizers
   d. Low pressure feedwater heaters
   e. Main feed pumps (MFPs)
   f. High pressure feedwater heaters
   g. Feedwater regulating and bypass valves
   h. Feedwater isolation valves (FWIVs)
   i. Steam generators
   j. Startup auxiliary feedwater (AFW) pump
   k. Heater drain system
   l. Condensate storage tank (CST).

2. List the components and connections located in the Seismic Category I portion of the feedwater system piping, and explain the purpose of each.

3. Explain how cascading heater drains increase plant efficiency.

7.3 WESTINGHOUSE TURBINE AND AUXILIARIES

Learning Objectives:

1. State the purposes of the turbine and turbine auxiliaries.

2. Identify the sources of heating steam to the moisture separator reheaters (MSRs).

3. State the purposes of the following turbine valves:

    a. Throttle (stop) valves,
    b. Governor (control) valves,
    c. Reheat stop valves, and
    d. Intercept valves.

8.1 ROD CONTROL SYSTEM

Learning Objectives:

1. State the purposes of the rod control system.

2. List the inputs to the automatic rod control system and explain why each is used.

3. Explain why the rate of change of the difference between turbine and nuclear power is used in the power mismatch circuit.

4. Explain why the gain of the power mismatch circuit is reduced at higher nuclear powers.

5. Given a list, arrange in proper order the stepping sequence of the Control Rod Drive Mechanism (CRDM).

6. List the rod withdrawal stops that occur in both automatic and manual rod control.

7. Describe the effects of an "urgent failure" in the logic cabinet and in a power cabinet.

8. Explain how individual rod motion is achieved.

9. Briefly describe the functions of the Bank Overlap Unit (BOU).

10. Briefly describe the rod speed program and explain the purpose of "deadband" and "lock-up".

8.2 ROD POSITION INDICATION (ANALOG)

Learning Objectives:

1. State the purpose of the rod position indication system.

2. Briefly describe the operation of the following:
   a. Analog Individual Rod Position Indication (IRPI), and
   b. Group demand position indication.

3. List three conditions that will initiate a rod deviation alarm.

8.4 ROD INSERTION LIMITS

Learning Objectives:

1. State the purposes of the control Rod Insertion Limits (RIL).
2. List the inputs to the rod insertion limit computer.
3. Explain why the rod insertion limits increase with increasing reactor power.

9.1 EXCORE NUCLEAR INSTRUMENTATION

Learning Objectives:

1. List the purposes of the excore nuclear instrumentation system.
2. List the reactor protection system inputs provided by the excore nuclear instrumentation system and the purpose (basis) of each.
3. List the interlocks and permissives provided by the excore nuclear instrumentation system and the purpose (basis) of each.
4. Explain how the excore nuclear instrumentation system is capable of detecting both axial and radial (azimuthal) power distribution.
5. Explain how the power range signal is calibrated to indicate reactor thermal output.
6. Explain why gamma compensation is required only in the source and intermediate ranges.
7. Explain the effects of an improperly compensated intermediate range.
8. Explain why channel test signals are additive to the channel outputs.

9.2 INCORE INSTRUMENTATION SYSTEM

Learning Objectives:

1. State the purposes of the incore instrumentation system.
2. Briefly describe the two types of incore instrumentation and the information available from each.
3. Describe the method used to detect flux thimble leakage.
4. Describe how a full-core flux map can be obtained using a limited number of detectors.
5. List four uses of the data obtained from the incore instrumentation.

10.1 REACTOR COOLANT INSTRUMENTATION

Learning Objectives:

1. Describe how loop average temperature T_{avg} and temperature difference ΔT are derived from the narrow range resistance temperature detector (RTD) outputs and how these signals are used.

2. List the functions of the following temperature monitors:

    a. Reactor Coolant System (RCS) wide range temperature detectors
    b. Pressurizer, surge and spray line detectors
    c. Safety and relief valve discharge line de-tectors
    d. Pressurizer relief tank (PRT) detectors
    e. Reactor vessel flange leakoff detector

3. Explain how the RCS piping elbow and differential pressure ΔP cells are used to develop the RCS flow signal.

10.2 PRESSURIZER PRESSURE CONTROL SYSTEM

Learning Objectives:

1. List and describe the purposes (bases) of the protective signals provided by the pressurizer pressure transmitters.

2. List and describe the purposes of the permissives and interlocks provided by the pressurizer pressure transmitters.

3. List in sequence the actions performed by the pressure control system during:

   a. A continuous pressure increase above 2235 psig
   b. A continuous pressure decrease below 2235 psig

4. Explain the effect of changing the pressure control setpoint on both control and protective functions.

5. List the inputs to the cold overpressure protection system and explain the operation of the system.

10.3 PRESSURIZER LEVEL CONTROL SYSTEM

Learning Objectives:

1. State the purposes of the pressurizer level control system.

2. List and describe the purposes (bases) of the protective signal provided by pressurizer level instrumentation.

3. Identify the instrumentation signal that is used to generate the pressurizer level program, and explain why level is programmed.

4. Explain how charging flow is controlled in response to pressurizer level error signals during the following:

   a. Centrifugal charging pump operation
   b. Positive displacement charging pump operation.

5. Explain the purposes of the pressurizer low level interlocks.

11.1 STEAM GENERATOR WATER LEVEL CONTROL SYSTEM

Learning Objectives:

1. State the purposes of the Steam Generator Water Level Control System.

2. List the Reactor Protection System and turbine trip inputs provided by steam generator instrumentation and state the purpose of each trip.

3. List the inputs used in the feedwater control system. Describe how and why each input is used.

4. List the inputs used in the main feed pump speed control system. Describe how and why each input is used.

11.2 STEAM DUMP CONTROL SYSTEM

Learning Objectives:

1. List the purposes of the steam dump system.
2. Briefly describe how each of the purposes is accomplished.
3. Discuss the inputs to the steam dump control system.
4. List and explain the purposes of the steam dump control arming signals and interlocks.

11.3 WESTINGHOUSE ELECTRO-HYDRAULIC CONTROL SYSTEM

Learning Objectives:

1. State the purposes of the turbine Electrohydraulic Control (EHC) System.

2. Describe the sequence of events which results in a turbine trip when initiated by:

   a. Mechanical input (example: low condenser vacuum)
   b. Electrical input (example: reactor protection signal).

3. Explain what function the following components perform in initiating a turbine trip:

   a. Solenoid trip mechanism
   b. Interface valve
   c. Emergency trip valve.

4. List the input parameters to the control systems used for the following turbine operational modes:

   a. Speed control
   b. Load control.

5. Explain the difference between "impulse in" and "impulse out" during turbine load control operations.

6. Briefly describe the actions initiated by the overspeed protection controller (OPC) for the following conditions:

   a. 103% turbine overspeed
   b. Opening of the generator output breakers at high loads.

7. Explain how the following types of turbine runbacks are accomplished:

   a. Secondary system initiated
   b. Reactor protection system initiated.

8. List the turbine trip indication inputs to the reactor protection system.

12.1 REACTOR PROTECTION SYSTEM

Learning Objectives:

1. State the purposes of the Reactor Protection System (RPS)

2. Explain how the following design features are incorporated into the RPS:
   a. Single failure criterion
   b. Testability
   c. Equipment qualification
   d. Independence
   e. Diversity
   f. Control and protection system interaction

3. Describe the sequence of events (flowpath) beginning at the sensor up to and including the starting of an Engineered Safety Feature (ESF) component and/or the opening of a reactor trip breaker.

4. Explain how failures in the rod control system are prevented from affecting reactor trip capability.

12.2 REACTOR PROTECTION SYSTEM REACTOR TRIP SIGNALS

Learning Objectives:

1. Given a list of reactor trips, explain the purpose (basis) of each.
2. Given a list of Reactor Protection System (RPS) interlocks, explain the purpose of each.
3. Given a list of control grade interlocks, explain the purpose of each.

12.3 REACTOR PROTECTION SYSTEM ENGINEERED SAFETY FEATURES ACTUATION SIGNALS

Learning Objectives:

1. List the Engineered Safety Features (ESF) actuation signals and the accident(s) which will initiate each.

2. List the systems or components that are actuated by an engineered safety features actuation signal.

3. Describe the effects of "resetting" an engineered safety features actuation signal and how the "reset" signal is automatically removed.

13.0 INSTRUMENT AND SERVICE AIR SYSTEMS

Learning Objectives:

1. Explain the purpose of the instrument and service air system.

2. Describe the basic components of the instrument and service air system.

3. Explain the interconnections between the instrument and service air system and support systems.

14.1 COMPONENT COOLING WATER SYSTEM

Learning Objectives:

1. State the purposes of the Component Cooling Water (CCW) System .

2. List the loads served by the CCW system.

3. Explain how the design of the CCW system prevents the release of radioactivity to the environment.

4. Describe both methods of detecting leakage into the CCW system.

5. Describe how the CCW system is protected against leakage in the thermal barrier heat exchangers.

14.2 SERVICE WATER SYSTEM

Learning Objectives:

1. State the purposes of the Service Water System (SWS).
2. List two nonsafety related loads supplied by SWS.
3. List four safety related loads supplied by SWS.

14.3 CIRCULATING WATER SYSTEM AND TURBINE BUILDING COOLING WATER SYSTEM

Learning Objective:

1. State the purposes of the circulating water system and turbine building cooling water system.

14.4 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

Learning Objectives:

1. State the purposes of the spent fuel pool cooling and cleanup system.

2. Describe the design features of the spent fuel pool cooling and cleanup system which prevent inadvertently lowering the water level in the spent fuel pool.

15.1 LIQUID RADIOACTIVE WASTE SYSTEMS

Learning Objectives:

1. Explain why liquid radioactive waste is separated into reactor grade and non-reactor grade.
2. List three inputs to the reactor grade waste subsystem.
3. List three categories of non-reactor grade wastes.
4. Briefly explain three liquid radioactive waste processing methods.
5. Define "decontamination factor" (D.F.).

15.2 SOLID RADIOACTIVE WASTE SYSTEMS

Learning Objectives:

1. List the two categories of solid radwaste.
2. Provide three examples of each category.
3. List and briefly describe two solidification systems.

15.3 GASEOUS RADIOACTIVE WASTE SYSTEMS

Learning Objectives:

1. List both the principle volume contributors and radioisotope contributors to the gaseous radioactive waste system.

2. Explain the basis for the curie limit placed on a gas decay tank.

3. Explain the reasons for using cover gas.

16.0 RADIATION MONITORING SYSTEM

Learning Objectives:

1. State the purposes of the radiation monitoring system.

2. List the two classes of radiation monitors and give four examples of each.

3. List four radiation monitors that provide automatic actions (other than alarms) and briefly describe the action provided.

4. List and briefly describe the two types of failed fuel monitors.

5. List the radiation monitors which identify the following:

   a. Primary to secondary leakage
   b. Primary to containment leakage

17.1 FUEL HANDLING AND STORAGE

Learning Objectives:

1. State the purposes of the fuel handling and storage systems.

2. State the functions of the following fuel handling system equipment:

    a. Spent fuel pool bridge crane
    b. New fuel elevator
    c. Fuel transfer canal
    d. Polar crane
    e. Manipulator crane
    f. RCCA change fixture
    g. Reactor vessel stud tensioner
    h. Conveyor car
    i. Upenders

3. State the reasons for handling spent fuel under water.

19.0 PLANT OPERATIONS

Learning Objectives:

1. Given a list of plant evolutions similar to the following, arrange the list in the order that they are performed during plant start-up, shutdown, or power operations.

    a. Reactor coolaant system (RCS) fill and vent
    b. Establish pressurizer steam bubble
    c. Start reactor coolant pumps
    d. Place all engineered safety features (ESF) systems in an operable condition
    e. Establish no-load T_{avg}
    f. Take reactor critical
    g. Place main feedwater in service
    h. Synchronize and load turbine generator
    i. Place rod control system in auto, place steam generator level control in auto, shift steam dump system to T_{avg} mode
    j. Escalate turbine generator load to desired value.